DOCUMENT RESUME

ED 195 410 SE 033 401

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TITLE	Nuclear Engineering Computer Modules, Thermal-Hydraulics, TH-1: Pressurized Water Reactors.
INSTITUTION	Virginia Polytechnic Inst. and State Univ., Blacksburg. Coll. of Engineering.
SPONS AGENCY	National Science Foundation, Washington, D.C.
PUB DATE	[73]
GRANT	NSF-GZ-2888: NSF-SED73-06276
NOTE	95p.: For related documents, see SE 033 402-403. Contains marginal legibility in computer printouts.
EDRS PRICE	MF01/PC04 Plus Postage.
DESCRIPTORS	*College Science: *Computer Programs: Engineering: *Engineering Education: Fuels: Higher Education: Hydraulics: *Learning Modules: Mathematical Models: Mechanics (Physics): *Nuclear Physics: Physics: *Science Education: Science Instruction: Thermodynamics

ABSTRACT

This learning module is concerned with the temperature field, the heat transfer rates, and the coolant pressure drop in typical pressurized water reactor (PWR) fuel assemblies. As in all of the modules of this series, emphasis is placed on developing the theory and demonstrating its use with a simplified model. The heart of the module is the PWR Thermal-Hydraulics Computer Code which solves for the radial temperature distributions in the fuel, cladding, and coolant of any axial station and then marches axially with an energy balance in the coolant. The code and its use are described in detail, including a listing and definition of all variables, a discussion of all input requirements and resulting output, an annotated flow-chart of the code, an explanation of all options in the code, and a listing of the code which gives enough comment statements to clearly indicate the operational steps being performed. By proper specification of the options, the code can either be used as an individual entity to study thermal-hydraulic aspects exclusively, or as a subroutine in the total PWR module package to provide temperature feedback to the other modules. Examples are worked out using the code. (Author/SK)

THERMAL-HYDRAULICS MODULE, TH-1

PRESSURIZED WATER REACTOR THERMAL-HYDRAULICS

by

Thomas C. Reihman

The University gratefully acknowledges the support of the Division of Higher Education of the National Science Foundation for support of this work performed under Grant GZ-2888 and the support of Duke Power Company, North Carolina Power and Light Company, and Virginia Electric and Power Company.

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PRESSURIZED WATER REACTOR THERMAL-HYDRAULICS

1.0 Object of Module

The object of this module is to present the basic [illegible] of pressurized water reactor thermal-hydraulics. This r[illegible] demonstration of:

- How the actual fuel geometry can be modeled for simplified thermal-hydraulic analysis.
- What information is necessary to characterize the thermal-hydraulic behavior of the reactor.
- The development of the theoretical relations that permit the computation of these thermal-hydraulic characteristics.
- The actual calculation of this information for the reactor. This calculation requires the use of the PWR Thermal-Hydraulics Code, the description of which is included in this module.

The thermal-hydraulic characteristics of the reactor are required for the determination of:

- Fuel integrity
- Cladding integrity
- Coolant exit conditions
- Pumping requirements
- Temperature feedback for reactor neutronics calculations.

2.0 Content of the Module

This learning module contains the thermal-hydraulics of pressurized water reactors. Specifically, the module is concerned with the temperature field, the heat transfer rates and the coolant pressure drop in typical pressurized water reactor fuel assemblies.

As in all of the modules of this series, emphasis is placed on developing the theory and demonstrating its use with a simplified model. The model is carefully selected to insure that analyses based on it will exhibit all of the important thermal-hydraulic trends of the typical reactor. The description of the fuel for a typical pressurized water reactor and the modeling of the fuel which preserves the significant thermal-hydraulic characteristics are treated in the next section of this module.

Following the geometry and modeling section, the basic theory governing the temperature distributions, heat transfer rates, pressure drops, and energy balance considerations is presented. The temperature distributions in the fuel and cladding are calculated assuming one-dimensional radial heat conduction. The pressure drop in the coolant channels and the heat transfer coefficient for use in Newton's law of cooling are calculated from empirical relations developed for reactor coolant channel flows. Energy balances for small axial segments of the coolant channel are used to step the solution in the axial direction. Simple examples, illustrating the individual calculations, are worked out in detail for typical PWR conditions.

The heart of the module is the PWR Thermal-Hydraulics Computer Code. Basically, the code solves for the radial temperature distributions in the fuel, cladding, and coolant at any axial station and then marches axially with an energy balance in the coolant. The code and its use are described in detail. Included are a listing and definition of all variables, a discussion of all input requirements and resulting output, an annotated flow chart of the code, an explanation of all options in the code, and a listing of the code which includes enough comment statements to clearly indicate the operational steps being performed. By proper specification of the options, the code can either be used as an individual entity to study thermal-hydraulic aspects exclusively, or as a subroutine in the total PWR module package to provide temperature feedback to the other modules. Examples are worked out using the code. In typical examples, the location and magnitude of the maximum fuel temperature in the PWR are found and the effect of oversized cladding on maximum fuel temperature and coolant outlet temperature are determined.

3.0 PWR Fuel Geometry and Its Model

The typical pressurized water reactor core is approximately cylindrical in shape and is composed of 200 to 240 fuel assemblies. These fuel assemblies are rectangular in cross section, 12 to 14 feet long, and stand vertically in the core. Water gaps (the order of 0.20 in.) separate the fuel assemblies from one another. The fuel assemblies are contained inside a core shroud which in turn fits inside a cylindrical core support which is 10 to 12 feet in diameter. To provide a flattening of the neutron flux (and hence the power production) in the radial direction, the enrichment of the fuel is varied in cylindrical zones.

The fuel assembly consists of 150 to 200 fuel rods, support grids holding the rods firmly at the ends, and spacers between the individual rods at several axial locations. All but about 1 foot of the 12 to 14 feet length of the fuel rods contain nuclear fuel and therefore generate heat internally during reactor operation.

A cross-section of a typical fuel assembly is shown in Figure 1. The fuel rods are arranged in a square array within the fuel assembly. The coolant, which carries away the energy generated within the fuel rods, flows in the spaces between the rods. Three types of coolant channels can be identified as shown in Figure 1. These correspond to the channels associated with central fuel rods, outer row fuel rods, and corner fuel rods.

Noting that a great majority (144 out of 196 in a 14 by 14 array) of the coolant channels are central-type coolant channels, and that the outer row- and corner-type channels do not differ too markedly from the central-type (elongated about 10% in one or both directions, respectively) one of

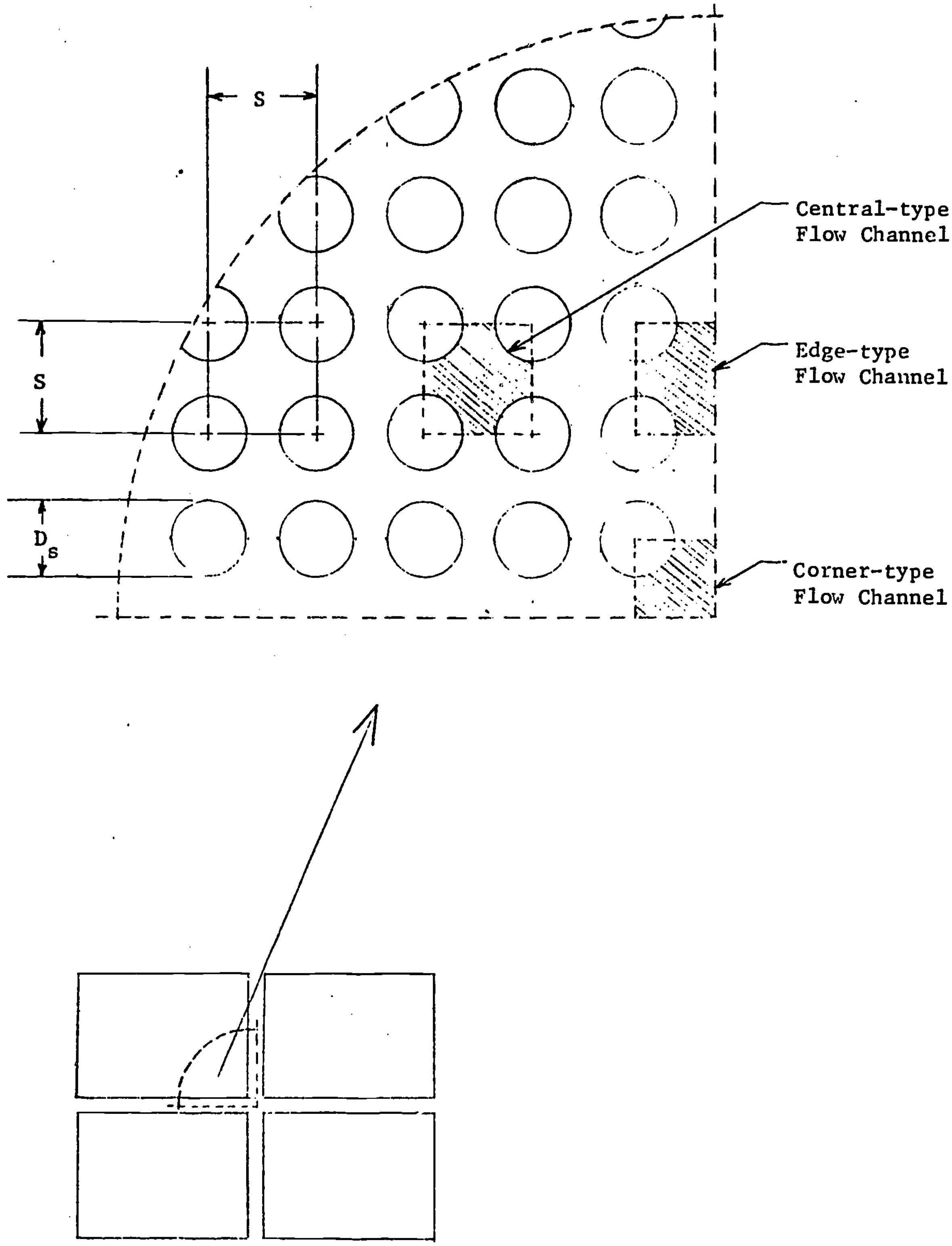


Figure 1. PWR Fuel Geometry.

the central-type channels is selected as the model coolant channel. The model channel thus consists of a square with corners at the centers of the four fuel rods forming the channel minus the four quarter-circular segments of the fuel rods themselves. The ratio of center-to-center spacing, S, to the fuel rod diameter, D_s, is typically 1.3. The hatched area in Figure 1 shows a model channel.

Each flow channel serves segments of four fuel rods as can be seen in Figure 1. Since the four rods are identical and since the neutron flux does not vary significantly in the short distance from one rod to the next all four rods may be assumed to have the same operating characteristics. Thus the four fuel rod segments associated with each flow channel can be treated as one equivalent fuel rod. The power generation level in the model fuel rod will be taken as that of the average fuel rod in the core.

The model fuel rod will have a variation in power density in the axial direction. This variation will be taken as

$$q'''(Z) = q'''_o \cos \frac{\pi Z}{H_e} \qquad (1)$$

unless a computed actual axial variation is provided from another module. In Equation 1, q''' represents the thermal source strength per unit volume at any axial location Z, q'''_o represents the thermal source strength per unit volume at the center of the fuel rod (Z=0), and H_e is the extrapolated height of the core. Both q''' and q'''_o are taken to be constant radially throughout the fuel in the fuel rod. The magnitude of q'''_o is representative of that of the average fuel rod in the core. The extrapolated height is calculated from

$$H_e = H + 2\,L_e \qquad (2)$$

where H is the actual height of the reactor core and L_e is an extrapolation length. The extrapolation length is the distance between the actual end of the core and the location where an extrapolation of the waveform representing the actual neutron flux distribution within the core goes to zero. From neutron diffusion theory L_e can be shown to be about one migration length for a bare core. The migration length can be calculated from neutron diffusion theory [1,2,3]*.

The PWR fuel rods are 12 to 14 feet long and about 0.45 inch in diameter. About 8 to 10 inches of the total length is used for end fittings and a fission gas plenum. The remaining length contains nuclear fuel, usually uranium dioxide. The fuel is typically in the form of cylindrical pellets with a diameter of about 0.39 inch. These fuel pellets are stacked inside a tube of cladding, usually a zirconium alloy with about a 0.025 inch thick wall. The 0.003 to 0.005 inch gap between the fuel and the inside of the cladding is pressurized with high-purity helium.

For simplicity, the helium gap is neglected in the model of the fuel rod. The model fuel rod thus consists of fuel of diameter D_f contained in a tube of cladding with an inside diameter D_f and an outside diameter D_s. For the model fuel rod, the length of the end fittings will each be assumed to be 1% of the active fuel length and the length of the gas plenum on the top of the fuel will be taken as 5% of the active fuel length.

*Numbers in brackets refer to items in References.

4.0 PWR Thermal-Hydraulic Theory

4.1 Internal Heat Generation

As a result of nuclear fission in the fuel, heat is generated. The rate of energy generation in the fuel per unit volume is called the "volumetric thermal source strength," q''', and can be calculated from

$$\dot{q}''' = E_f N_{ff} \bar{\sigma}_f \phi \tag{3}$$

where E_f is the energy released per fission reaction (energy dimensions), N_{ff} is the fissionable fuel density (fissionable nuclei per unit volume), $\bar{\sigma}_f$ is the effective fission microscopic cross section (dimensions of area), and ϕ is the neutron flux (neutrons per unit area per unit time). Note that q''' has dimensions of energy per unit volume. The details of this calculation of volumetric thermal source strength are found in Reactor Statics Module 8.

The volumetric thermal source strength varies throughout the reactor since ϕ, and perhaps N_{ff}, vary. For a cylindrical reactor, axial symmetry is generally a reasonable approximation. Therefore q''' reduces to a function of only the radius, R, and the axial position, Z. Across any single fuel rod, the change in q''' is small (due to small change in R) and q''' can therefore be considered constant across its cross section. Thus for any single fuel rod q''' is a function of only the axial position. Also for any small axial segment of a fuel rod (the order of 1% of the total active length) the change in q''' is moderate. Using a constant average value for any such segment thus introduces little error into the analysis. This approximation is used in this module.

The heat generation in any small segment of the fuel can be obtained by multiplying the volumetric thermal source strength by the volume of the segment. For steady state conditions this energy must be removed from the fuel. The mechanism by which this heat transfer occurs within the fuel is thermal conduction.

Example 1

A PWR containing 25,000 fuel rods generates 2000 Mw(t). The cylindrical core of the reactor is 10 ft long and 7 ft in diameter. The fuel consists of 0.3 in. diameter UO_2 fuel pellets contained in cladding which has an outside diameter of 0.35 in. The fuel rods are spaced 0.45 in. center-to-center in a square array. The volumetric thermal source strength of the fuel varies as $q''' = q_o''' \cos(\Pi Z/H_e)$. The extrapolation length for the fuel is 6 in. Find the power per unit length (in kw/ft) and q_o''' for the average fuel rod in the core.

Solution

The power generated in the average fuel rod is

$$P_{ave} = \frac{P\ total}{N} = \frac{2000 \times 10^6}{25,000} = 80,000 \text{ watt} = 80 \text{ kw}.$$

Thus, per unit length the power is $\frac{P}{L} = \frac{80 \text{ kw}}{10 \text{ ft}} = 8 \frac{\text{kw}}{\text{ft}}$.

The power produced by the average fuel rod can also be calculated by integrating the power produced in each differential volume of the fuel rod. Thus,

$$P_{ave} = \int_V q''' dV = \int_{H/2}^{H/2} q_o''' \cos(\Pi Z/H_e)(\Pi D_f^2/4)dZ$$

$$= (H_e D_f^2 q_o'''/4)[\sin(\Pi Z/H_e)]_{-H/2}^{H/2}$$

$$= (H_e D_f^2 q_o'''/2) \sin(\Pi H/2H_e).$$

Solving for q_o''':

$$q_o''' = \frac{2P_{ave}}{H_e D_f^2 \sin(\pi H/2H_e)}$$

$$= \frac{(2)(80{,}000 \text{ watt})(3.413 \frac{\text{Btu}}{\text{watt-hr}})}{(11 \text{ ft})(0.3 \text{ in.})^2 (\frac{1}{12} \frac{\text{ft}}{\text{in.}})^2 \sin[(\pi)(10 \text{ ft})/(2)(11 \text{ ft})]}$$

$$= 8.025 \times 10^7 \text{ Btu/hr-ft}^3.$$

4.2 Temperature Distribution in the Fuel

For conduction heat transfer the heat flux is proportional to the normal temperature gradient. When the proportionality constant is inserted

$$q_n = -kA_n \frac{\partial T}{\partial X_n} \tag{4}$$

where q_n is the heat transfer rate in the n-direction (energy/time), $\partial T/\partial X_n$ is the temperature gradient in the n-direction (degrees/length), A_n is the area normal to the n-direction, and k is the thermal conductivity of the material (energy/degree-length-time). Equation 4, which is Fourier's law of heat conduction, relates the heat transfer to the temperature field and is also the defining equation for the thermal conductivity. The thermal conductivity is a material property and its magnitude in general varies with the temperature of the material. Heat transfer properties of various reactor materials are tabulated as functions of temperature in References 4, 5, and 6. To reduce the complexity of heat transfer calculations, the thermal conductivity is often assumed to be constant and evaluated at an average temperature. The minus sign in Equation 4 assures that the heat transfer is in the direction of decreasing temperature. Equation 4 shows that temperature gradients are required for heat transfer. In nuclear reactor applications, where there are high heat transfer rates, large temperature variations occur. One of the primary tasks of reactor thermal-hydraulic analysis is the prediction of this temperature distribution in the fuel.

The heat conduction equation in cylindrical coordinates

$$\frac{\partial^2 T}{\partial r^2} + \frac{1}{r}\frac{\partial T}{\partial r} + \frac{1}{r^2}\frac{\partial^2 T}{\partial \theta^2} + \frac{\partial^2 T}{\partial Z^2} + \frac{q'''}{k} = \frac{1}{\alpha}\frac{\partial T}{\partial T} \tag{5}$$

and the initial and boundary conditions prescribe the temperature distribution within the fuel rod. In this relation, the thermal conductivity, k, and the thermal diffusivity, α, have been assumed constant. The development of this equation is found in heat transfer texts [4, 5, 6] and several simplified cases are given as exercises for the student. For steady state conditions, the unsteady term on the right hand side of the equation is zero, and for axial symmetry, the Θ variation disappears. It may also be observed that since the length of a fuel pin is much greater than its radius, the temperature gradients in the radial direction will be much greater than the temperature gradients in the axial direction. Therefore, to a good approximation, the heat transfer in the axial direction can be neglected with respect to that in the radial direction, and the resulting temperature distribution and heat transfer reduced to a one-dimensional case for any axial segment in which q''' may be assumed constant. The governing differential equation for this case reduces to the ordinary differential equation

$$\frac{d^2T}{dr^2} + \frac{1}{r}\frac{dT}{dr} + \frac{q'''}{k_f} = 0. \tag{6}$$

The solution of Equation 6 subject to the boundary conditions

$$T = T_o \text{ at } r = 0, \qquad \frac{dT}{dr} = 0 \text{ at } r = 0 \tag{7}$$

yields the temperature distribution in the fuel. The second boundary condition is obtained from the observation that the temperature distribution must be continuous across the center of the cylinder. Observing that both q''' and k are constant and that

$$\frac{d^2T}{dr^2} + \frac{1}{r}\frac{dT}{dr} = \frac{1}{r}\frac{d}{dr}\left(r\frac{dT}{dr}\right) \tag{8}$$

Equation 6 can be written as

$$\frac{d}{dr}\left(r\ \frac{dT}{dr}\right) = -\ \frac{q'''}{k_f}\ r. \tag{9}$$

Integrating twice results in

$$T = -\frac{q'''}{k_f}\ \frac{r^2}{4} + C_1 \ln r + C_2. \tag{10}$$

Applying the boundary conditions of Equation 7, the integration constants are

$$\begin{aligned} C_1 &= 0, \\ C_2 &= T_o. \end{aligned} \tag{11}$$

Substituting into Equation 10 gives

$$T = T_o - \frac{q'''}{4k_f}\ r^2. \tag{12}$$

This relation shows the temperature distribution in the fuel to be parabolic with maximum temperature at the center. The heat transfer rate through any cylindrical shell can be calculated from Fourier's law which takes the form

$$q_r = -\ k_f\ A_r\ \frac{dT}{dr} \tag{13}$$

where $A_r = 2\pi r(\Delta L)$, ΔL being the length of the cylindrical shell.

Of particular interest are the temperature and heat transfer at the surface of the fuel; i.e., at $r = r_f = D_f/2$. At this location

$$T_f = T_o - \frac{q'''}{4k_f}\ r_f{}^2, \tag{14}$$

$$q_f = -k_f 2\pi r_f \, (\Delta L) \left.\frac{dT}{dr}\right|_{r = r_f}$$

$$= \pi r_f^2 (\Delta L) \, q''' . \tag{15}$$

Noting that $\pi r_f^2 \, \Delta L$ is the volume of the fuel rod segment of length ΔL, it is observed that the heat transfer out of the $r = r_f$ cylindrical shell is indeed equal to the total energy generated as calculated from (q''') (fuel volume). For one-dimensional heat transfer all of the energy generated within the fuel must be transferred out through the surface.

Example 2

Equation 14 relates the temperatures at the center and surface of a heat-generating cylindrical fuel rod. This equation contains the thermal conductivity of the fuel which, in general, is a function of temperature. However, the analysis leading to Equation 14 assumes the thermal conductivity to be constant. Determine the temperature at which to evaluate k_f to make the assumption of constant k_f compatible with a k_f that varies linearly with temperature.

Solution

The method of solution is to develop an expression relating T_o and T_s which assumes $k_f = a + bT$ and then compare this result with Equation 14. We begin with Fourier's law of heat conduction,

$$q = -k_f A_r \frac{dT}{dr}.$$

Recognizing that $q = q'''$ (volume) and $A_r = 2\pi r(\Delta L)$:

$$q''' \, \pi r^2 (\Delta L) = -k_f \, 2\pi r (\Delta L) \frac{dT}{dr}.$$

Simplifying, separating variables, and substituting for k_f:

$$q''' \, r dr = -2(a + bT)dT.$$

Integrating from $r = 0$ to $r = r_f$ with q''', a, b constant,

$$q''' \left[\frac{r^2}{2}\right]_0^{r_f} = -2\left[aT + \frac{b}{2}T^2\right]_{T_o}^{T_f}$$

$$q''' \frac{r_f^2}{4} = -a(T_f - T_o) - \frac{b}{2}(T_f^2 - T_o)^2$$

$$= [a + \frac{b}{2}(T_o + T_f)]\,(T_o - T_f)$$

$$T_o - T_f = \frac{q''' \, r_f^2}{4[a + \frac{b}{2}(T_o + T_f)]}.$$

Comparing this with Equation 14 written as

$$T_o - T_f = \frac{q''' \, r_f^2}{4\, k_f}$$

we see that the relations are equivalent for

$$k_f = a + \frac{b}{2}(T_o + T_f).$$

Note that this is precisely equal to k_f evaluated at the arithmetic average fuel temperature; i.e., $k(\frac{T_o + T_f}{2})$.

4.3 Temperature Distribution in the Cladding

The temperature distribution in the cladding can be calculated by methods similar to those used for the fuel. Noting that no energy is generated in the fuel, the governing equation simplifies to

$$\frac{d}{dr}\left(r\,\frac{dT}{dr}\right) = 0. \tag{16}$$

The boundary conditions obtained by matching temperature and heat transfer at the fuel-cladding interface are

$$T = T_f \text{ at } r = r_f,$$

$$\frac{dT}{dr} = -\frac{q_f}{2\pi k_c r_f(\Delta L)} = -\frac{r_f q'''}{2k_c} \quad \text{at } r = r_f \tag{17}$$

where k_c is the thermal conductivity of the cladding.
Integrating Equation 16 twice yields

$$T = C_3 \ln r + C_4. \tag{18}$$

Applying the boundary conditions in Equation 17,

$$C_3 = -\frac{r_f^2 q'''}{2k_c},$$

$$C_4 = T_f + \frac{r_f^2 q'''}{2k_c} \ln r_f. \tag{19}$$

Substituting into Equation 18 yields the temperature distribution in the cladding,

$$T = T_f - \frac{r_f^2 q'''}{2k_c} \ln \frac{r}{r_f}. \tag{20}$$

Or, if q''' is eliminated in favor of q_f,

$$T = T_f - \frac{q_f}{2\pi k_c(\Delta L)} \ln \frac{r}{r_f}. \tag{21}$$

The temperature at the surface of the cladding, T_s, is obtained by evaluating Equation 21 at r_s,

$$T_s = T_f - \frac{q_f}{2\pi k_c(\Delta L)} \quad \ln \frac{r_s}{r_f} . \tag{22}$$

Note also that the same heat transfer rate must occur through each cylindrical layer for steady state conditions; i.e., $q_f = q_s$.

4.4 Temperature Drop in the Convective Layer

All of the heat generated within the fuel ultimately must be transferred to the coolant. Thus all of this energy must be transferred through the fluid layer near the outer surface of the cladding. The heat transfer mechanism, wherein the energy is carried away or convected from the solid surface by a fluid in motion, is called convection. The heat transfer rate for convective heat transfer is related to the temperature difference between the surface and the bulk fluid, the driving potential for the heat transfer, by Newton's law of cooling,

$$q_s = h\, A_s\, (T_s - T_B). \tag{23}$$

This relation may also be taken as the defining equation for the heat transfer coefficient, h. The bulk temperature T_B, is a mass-weighted average temperature of the fluid in the flow channel. It is formally defined by

$$T_B = \frac{\int_A \rho C_p U T \, dA}{\dot{m} C_p}. \tag{24}$$

This is the temperature that a thermometer would indicate if immersed in a cup of fluid collected from the discharge of the flow channel at the given location.

Relations for calculating h from the flow characteristics and coolant properties have been developed. These will be treated in a later section of the module. For the moment, the quantity of interest is the temperature drop across the convective layer. In terms of h, this can be obtained from Equation 23 as

$$T_s - T_B = \frac{q_s}{2\pi r_s(\Delta L)h} \qquad (25)$$

where r_s is the outer radius of the cladding.

Example 3.

Consider the PWR described in Example 1. The bulk temperature of the coolant changes from 491 F to 609 F as the coolant flows through the reactor. Given that the heat transfer coefficient is 6000 Btu/hr-ft^2-F, the thermal conductivity of the fuel and cladding are 2.40 and 6.90 Btu/hr-ft-F, respectively, determine the fuel centerline temperature at the axial center of the core.

Solution.

Half of the total energy transfer to the coolant occurs in the first half of the core. Thus the coolant bulk temperature at the axial center of the core is 550 F. To this, we must add the temperature rises across the convective layer, the cladding, and the fuel to obtain the fuel centerline temperature.

The heat transfer rate through the convective layer at Z = 0 is

$$q_s = q_o''' \text{ (fuel volume)} = q_o''' \, \pi D_f^2 (\Delta L)/4$$

$$= (8.025 \times 10^7 \text{ Btu/hr-ft}^3)(\Pi/4)(0.3 \text{ in.})^2(\text{ft}/12 \text{ in.})^2(\Delta L \text{ ft})$$

$$= 39393\ (\Delta L) \text{ Btu/hr}.$$

From Equation 23,

$$T_s - T_B = \frac{q_s}{h\,A_s} = \frac{39393\ \Delta L \text{ Btu/hr}}{(6000 \text{ Btu/hr-ft}^2\text{-F})\Pi(0.35 \text{ in.})(\text{ft}/12 \text{ in.})(\Delta L \text{ ft})}$$

$$= 71.6 \text{ F}.$$

The heat transfer rate through the cladding is the same as that through the convective layer. Equation 22 yields

$$T_f - T_s = \frac{q_f}{2\pi k_c(\Delta L)} \ln \frac{r_s}{r_f}$$

$$= \frac{(39393\ \Delta L\ \text{Btu/hr})\ \ln(0.175/0.15)}{2\pi(6.90\ \text{Btu/hr-ft-F})\ (\Delta L\ \ \text{ft})}$$

$$= 140.1\ \text{F.}$$

From equation 14 the temperature rise through the fuel is found as

$$T_o - T_f = \frac{q''' \ r_f^2}{4\ k_f} = \frac{(8.025 \times 10^7\ \text{Btu/hr-ft}^3)(0.15\ \text{in.})^2(\text{ft}/12\ \text{in.})^2}{4(2.40\ \text{Btu/hr-ft-F})}$$

$$= 1306.2\ \text{F.}$$

Adding these temperature rises to the bulk temperature,

$$T_o = (T_o - T_f) + (T_f - T_s) + (T_s - T_B) + T_B$$

$$= 1306.2 + 140.1 + 71.6 + 550 = 2067.9\ \text{F.}$$

4.5 Pressure Drop in the Coolant

All of the energy generated by nuclear fission in the fuel must be carried out of the reactor by the coolant which flows in the channels between the fuel rods. In a PWR the coolant is water which is force-circulated by a pump. Enough pressure head must be provided by the pump to overcome the pressure losses in the water flow loop. The entire loop consists of the reactor flow channels, the heat exchangers, water quality maintenance components and connecting piping. Of interest here is the determination of the pressure loss incurred in the flow through the reactor core. Two types of pressure losses will be considered. These are the frictional pressure loss along the coolant channel and the entrance and exit permanent pressure losses.

The frictional pressure loss is a manifestation of the shear stress on the flowing fluid by the walls of the flow channel. This pressure loss is calculated from

$$\Delta P_F = f \frac{\Delta L}{D_e} \frac{\rho U_B^2}{2g_c} . \qquad (26)$$

This relation may also be interpreted as the defining equation for the friction factor, f. The bulk velocity, U_B, found in Equation 26 is the velocity averaged across the flow channel. It is defined by

$$U_B = \frac{1}{A} \int_A U \, dA. \qquad (27)$$

This velocity is related to the mass flow rate by

$$\dot{m} = \rho A U_B \qquad (28)$$

or to the mass velocity by

$$G = \frac{\dot{m}}{A} = \rho U_B. \qquad (29)$$

The equivalent diameter, D_e, is used to relate experimental data from circular tubes, which is abundant in the literature, to different geometries. The equivalent diameter is calculated from

$$D_e = \frac{4A_f}{p_w}. \qquad (30)$$

Note that for a circular cross section the flow area is $\Pi D^2/4$ and $p_w = \Pi D$. Thus for a circle the equivalent diameter is equal to the actual diameter, as required. This, of course, is why the definition of Equation 30 is used.

The value of the friction factor depends on the flow and surface conditions in the channel. The flow conditions are characterized by the Reynolds number,

$$Re = \frac{\rho U_B D_e}{\mu}. \qquad (31)$$

It is well known that for Re below a critical value the flow is laminar, and for Re above the critical value the flow is turbulent. For internal flows this critical Re is about 2000. Laminar flow may be thought of as an ordered process in which fluid layers slide over one another, being retarded only by the molecular interaction between the layers. The viscosity of the fluid quantifies the magnitude of this interaction. In laminar flow any disturbance in the fluid is damped by the viscous action. In turbulent flow there is an additional random transport mechanism operable. This mechanism may be modeled as eddies (finite sized patches of fluid which retain their characteristics for finite times) moving throughout the fluid, transporting mass, momentum, and energy by virtue of their movement. This action is quite violent and results in transport rates much greater than those by purely molecular activity in laminar flow. In turbulent flow, disturbances in the flow field grow and propagate resulting in increased turbulence levels downstream of the disturbances. Even in highly turbulent flows, there exists a layer near the solid bounding surfaces where the presence of the wall retards the penetration of eddies and therefore acts as a flow stabilizer. This results in a laminar sub-layer existing near any solid boundary. Even though this layer is very thin, it is very important, since much of the temperature and velocity change between wall and bulk conditions occurs in this layer. In reactor coolant channels the flow is highly turbulent with Re of the order of 100,000 and greater being quite common.

The surface condition of the flow channel is characterized by the ratio of the surface roughness height to the equivalent diameter of the flow channel, ε/D_e. The flow channel exhibits smooth tube behavior if the roughness height is less than the thickness of the laminar sublayer. For reactor coolant flow channels the surface conditions are controlled so that this criterion is met.

The calculation of the friction factor for the reactor coolant flow channels is a two-step procedure. First, the smooth circular tube friction factor is determined, and then a correction factor is applied to account for the actual channel shape. The smooth circular tube friction factor can be determined from a Moody chart where f_{circ} is plotted versus Re. These charts are found in most fluid mechanics texts and handbooks [7, 8, 9, 10]. An alternate method is to calculate the friction factor from a correlation in equation form. One of the most widely accepted for reactor work is

$$f_{circ} = \frac{0.184}{Re^{0.2}} . \tag{32}$$

This relation, like the Moody diagram, was obtained from a curve fit of experimental data in smooth circular tubes. The friction factor defined in this module is the Darcy-Weisbach friction factor. Care must be taken not to confuse this fraction factor with the Fanning friction factor ($f_{Fan} = f_{D-W}/4$) also found in the literature. A correction of the circular tube data to the rod bundle geometry has been obtained by Deissler and Taylor [11]. Their results, expressing f/f_{circ} as a function of S/D_s, can be approximated by the following polynomial fit:

$$f/f_{circ} = -3.475 + 8.053 \frac{S}{D_s} - 4.705 \left(\frac{S}{D_s}\right)^2 + 0.9162 \left(\frac{S}{D_s}\right)^3 . \tag{33}$$

The permanent pressure losses at the inlet and exit are the result of increased viscous energy dissipation resulting from increased turbulent activity. These losses may be calculated from

$$\Delta P_E = K \frac{\rho U_B^2}{2g_c} . \tag{34}$$

In this relation, K is the resistance coefficient for the inlet or exit geometry. The inlet may be approximated as a sudden contraction from a very large diameter to the equivalent diameter of the flow channel. For this sudden contraction, the resistance coefficient is 0.5 [10]. Similarly the exit may be approximated as a sudden expansion from D_e to a very large diameter. For this expansion, K is 1.0 [10].

Example 4.

The PWR described in Example 1 has an inlet coolant temperature of 491 F, an inlet coolant pressure of 2250 psia, and a coolant mass velocity of 2.5×10^6 lbm/hr-ft^2. The coolant exit temperature is 609 F. Estimate the frictional pressure loss in the core. Neglect entrance and exit losses.

Solution.

The coolant properties are evaluated at the average coolant temperature. At 550 F, 2250 psia:

$$\mu = 0.2336 \quad \text{lbm/hr-ft}$$

$$\rho = 46.47 \quad \text{lbm/ft}^3.$$

The average coolant velocity is found from

$$U_B = G/\rho = (2.5 \times 10^6 \text{ lbm/hr-ft}^2)/(46.47 \text{ lbm/ft}^3) = 53{,}800 \text{ ft/hr.}$$

The equivalent diameter is computed next.

$$A_f = S^2 - \frac{\pi}{4} D_s^2$$

$$= (0.45 \text{ in.})^2 - \frac{\pi}{4}(0.35 \text{ in.})^2 = 0.1063 \text{ in.}^2$$

$$p_w = \Pi D_s = \Pi(0.35 \text{ in.}) = 1.100 \text{ in.}$$

$$D_e = 4\, A_f/p_w = 4(0.1063 \text{ in.}^2)/(1.100 \text{ in.})$$

$$= 0.386 \text{ in.} = 0.0322 \text{ ft.}$$

The Reynolds number is

$$Re = U_B\, D_e\, \rho/\mu$$

$$= (53{,}800 \text{ ft/hr})(0.0322 \text{ ft})(46.47 \text{ lbm/ft}^3)/(0.2336 \text{ lbm/hr-ft})$$

$$= 344{,}600.$$

Using a circular tube friction factor correlation,

$$f_{circ} = 0.184/Re^{0.2} = 0.184/(344,600)^{0.2}$$

$$= 0.0144.$$

Applying the correction for the actual reactor flow channel,

$$f = f_{circ} [- 3.475 + 8.053(S/D_s) - 4.705(S/D_s)^2 + 0.916(S/D_s)^3]$$

$$= (0.0144) [- 3.475 + 8.053 (\frac{0.45}{0.35}) - 4.705 (\frac{0.45}{0.35})^2 + 0.916(\frac{0.45}{0.35})^3]$$

$$= 0.0151.$$

From the defining equation for the friction factor,

$$\Delta P_F = f(\Delta L/D_e) \rho U_B^2/2g_c$$

$$= (0.0151)(120 \text{ in.}/0.386 \text{ in.})(46.47 \text{ lbm/ft}^3)(53,800 \text{ ft/hr})^2$$

$$/(3600 \text{ sec/hr})^2 (2) (32.2 \text{ lbm-ft/sec}^2 - \text{lbf})$$

$$= 756.5 \text{ lbf/ft}^2$$

$$= 5.25 \text{ psi.}$$

Example 5

Each coolant channel in a fuel assembly has the same pressure drop. If all coolant channels are not identical the flow will redistribute such that this equal pressure drop is attained. Determine the relation between average velocity in the coolant channel (or equivalently the mass velocity) and the equivalent diameter of the channel which governs the flow distribution.

Solution.

The frictional pressure drop is given by

$$\Delta P = f \frac{\Delta L}{D_e} \frac{\rho U_B^2}{2g_c}.$$

Substituting for the friction factor,

$$\Delta P = \frac{0.184}{R_e^{0.2}} \frac{\Delta L}{D_e} \frac{\rho U_B^2}{2g_c}$$

$$= \frac{0.184}{\left(\frac{U_B D_e \rho}{\mu}\right)^{0.2}} \frac{\Delta L}{D_e} \frac{\rho U_B^2}{2g_c}$$

$$\alpha \quad U_B^{1.8}/D_e^{1.2}$$

if constant density and viscosity are assumed. For channels 1 and 2, each having equal pressure drop,

$$U_{B1}^{1.8}/D_{e1}^{1.2} = U_{B2}^{1.8}/D_{e2}^{1.2}$$

or

$$U_{B2}/U_{B1} = (D_{e2}/D_{e1})^{2/3}.$$

4.6 The Heat Transfer Coefficient

The heat transfer coefficient for reactor coolant channels is determined by a two-step procedure similar to that used previously for the friction factor. The heat transfer coefficient for flow in circular tubes is found first and then a correction is applied to obtain h for the actual reactor coolant channel geometry.

No direct analytical methods for predicting h for turbulent flow in circular tubes exist. However, by the use of an analogy between heat and momentum transport, the dependence of h on the flow field characteristics, i.e., on the Re can be predicted. Then, by applying the results of a heat transfer analysis for laminar flow over a flat plate, an estimate of the Prandtl number dependence on heat transfer can be obtained. The Pr may be interpreted as a dimensionless modulus relating the temperature field in the fluid to the flow field. For Pr of unity, similar velocity and temperature profiles exist in the fluid. Finally, the predicted expression for h is compared to empirical correlations found to fit the available experimental data.

The heat transfer across a fluid layer in laminar flow may be calculated from Fourier's law,

$$q = - k A \frac{dT}{dy}, \tag{35}$$

where y is the direction normal to the fluid flow. Similarly, the shear stress in the fluid can be related to the velocity gradient by

$$\tau = \mu \frac{dU}{dy} \tag{36}$$

which defines the viscosity coefficient μ. These relations can be rearranged to yield

$$\frac{q}{\rho C_p A} = -\alpha \frac{dT}{dy} \tag{37}$$

and

$$\frac{\tau}{\rho} = \nu \frac{dU}{dy}. \tag{38}$$

In these relations, $\alpha = \frac{k}{\rho C_p}$ is the molecular diffusivity of heat and ν is the kinematic viscosity or the molecular diffusivity of momentum. We now postulate that the eddy transport of heat and momentum present in turbulent flow can be expressed in the same form. To this end, we define the eddy diffusivity of heat, ε_H, and the eddy diffusivity of momentum, ε_M, as the parameters, that, when multiplied by the appropriate gradient, yield the corresponding transport rate for turbulent flow. In general ε_M and ε_H vary throughout the flow field. For the combined molecular transport plus turbulent eddy transport,

$$\frac{q}{\rho C_p A} = -(\alpha + \varepsilon_H) \frac{dT}{dy} \tag{39}$$

and

$$\frac{\tau}{\rho} = (\nu + \varepsilon_M) \frac{dU}{dy}. \tag{40}$$

We now assume that heat and momentum are transported by analogous processes and at the same rate. This requires that $\alpha = \nu$ and $\varepsilon_H = \varepsilon_M$. Since the $Pr = \nu/\alpha$, the first condition is equivalent to $Pr = 1$. Similarly a turbulent Prandtl number may be defined as $Pr_t = \varepsilon_M/\varepsilon_H$. Therefore the second condition is equivalent to $Pr_t = 1$. The basic assumption we have made also implies that both q and τ vary in the same way across the field; i.e.,

$$\frac{q}{C_p A \tau} = \text{Constant} \equiv \frac{q_w}{C_p A_w \tau_w}, \tag{41}$$

if the constant is expressed in terms of quantities at the surface or wall.

Dividing Equation 39 by Equation 40 and using the assumption discussed above,

$$\frac{q_w}{C_p A_w \tau_w} \; dU = -dT. \tag{42}$$

Integrating this expression between wall conditions and fluid bulk conditions,

$$\frac{q_w}{C_p A_w \tau_w} \int_{U_w}^{U_B} dU = - \int_{T_w}^{T_B} dT. \tag{43}$$

Since $U_w = 0$, this reduces to

$$\frac{q_w U_B}{C_p A_w \tau_w} = T_w - T_B. \tag{44}$$

To introduce h into this expression, Newton's law of cooling (Equation 23) is used to give

$$h = \frac{q_w}{A_w(T_w - T_B)} \; . \tag{45}$$

Substituting Equation 45 into Equation 44,

$$h = \frac{C_p \tau_w}{U_B} \; . \tag{46}$$

The shear stress at the wall can be eliminated from this expression in favor of the friction factor. To do this, τ_w is first related to the pressure loss over the length of the pipe, ΔL, and then Equation 26 relating the pressure loss and the friction factor is used. Consider the fluid in a section of pipe of diameter, D, and length, ΔL. For fully developed flow, the sum of the forces acting in the flow direction must equal zero. The forces that must be considered are the pressure force on the upstream circular face, $\frac{\pi}{4} D^2 P$, the pressure force on the downstream face, $\frac{\pi}{4} D^2 (P-\Delta P)$, and the shear force along the periphery, $\Delta\pi D L\tau_w$. Summing these forces with proper regard as to their direction gives

$$\frac{\pi}{4} D^2 P = \pi D\Delta L\tau_w + \frac{\pi}{4} D^2 (P-\Delta P) \tag{47}$$

and

$$\tau_w = \frac{D}{4\Delta L} \Delta P .$$

Using Equation 26 for ΔP, this becomes

$$\begin{aligned} \tau_w &= \frac{D}{4\Delta L} f \frac{\Delta L}{D} \frac{\rho U_B^2}{2} \\ &= f \frac{\rho U_B^2}{8} . \end{aligned} \tag{48}$$

Substituting Equation 48 into Equation 47,

$$h = C_p \rho U_B \frac{f}{8} . \tag{49}$$

Introducing the Stanton Number, a convenient dimensionless grouping common in heat transfer work, Equation 49 becomes

$$St \equiv \frac{h}{\rho C_p U_B} = \frac{f}{8} . \tag{50}$$

Using Equation 32 for f,

$$St = 0.023 R_e^{-0.2} . \tag{51}$$

This result is often found expressed in terms of the Nusselt number, defined as

$$Nu = \frac{hD}{k} . \tag{52}$$

One common interpretation of the Nu is that it is the ratio of the actual convection heat transfer from a surface to the heat transfer assuming that only molecular conduction in a stagnant fluid is present. Noting that

$$St = \frac{h}{\rho C_p U_B} = \frac{(\frac{hD}{k})}{(\frac{C_p \mu}{k})\ (\frac{\rho D U_B}{\mu})} = \frac{Nu}{Pr\ Re}, \qquad (53)$$

we can rewrite Equation 51 as

$$Nu = 0.023\ Pr\ Re^{0.8}. \qquad (54)$$

The Pr dependence found in Equation 54 is not correct. One reason why this incorrect dependence appears is that Pr = 1 was assumed early in the development; therefore one would not expect the Pr dependence to be properly represented in the result. The analysis of laminar flow heat transfer from a flat plate (one of the few cases which can be treated analytically) yields a $Pr^{1/3}$ dependence in an expression for the Nusselt number. In practice it turns out that this $Pr^{1/3}$ dependence is reasonably accurate for the turbulent pipe flow case of interest here as well. In fact some of the empirical correlations found in the literature exhibit this $Pr^{1/3}$ dependence in the Nu(Pr, Re) relation. However, the relation that is the best known, most widely used, and is recommended here is the Dittus-Boelter Equation [12],

$$Nu = 0.023\ Re^{0.8}\ Pr^{n}. \qquad (55)$$

For the fluid in the tube being heated, the recommended value of n is 0.4 and for the fluid in the tube being cooled, the recommended value of n is 0.3. The former case is the one found in reactor work. In Equation 55 several temperature dependent fluid properties appear (k, ρ, μ, C_p). These properties are to be evaluated at the average bulk temperature of the fluid for the segment of tubing of interest.

The heat transfer coefficient for circular tubes calculated from Equation 55 is then related to the heat transfer coefficient for the reactor coolant channel. This is done by using the equivalent diameter of the coolant channel in the Nusselt number and Reynolds number in the circular tube Nusselt number correlation and then applying the correction of Deissler and Taylor [11]. The Deissler and Taylor correction is the same as that used for the friction factor; i.e. it may be approximated by

$$\frac{Nu}{Nu_{circ}} = -3.475 + 8.053 \frac{S}{D_s} - 4.705 \left(\frac{S}{D_s}\right)^2 + 0.9162 \left(\frac{S}{D_s}\right)^3 . \qquad (56)$$

The heat transfer coefficient determined from this relation is the average value around the periphery of the fuel rod.

<u>Example 6.</u>

Determine the average heat transfer coefficient for the PWR described in Examples 1 and 4.

<u>Solution</u>.

The average heat transfer coefficient corresponds closely to that calculated at the average coolant conditions; i.e., 550 F and 2250 psia. Additional fluid properties required for this calculation are

$$C_p = 1.258 \text{ Btu/lbm-F},$$
$$k = 0.3346 \text{ Btu/hr-ft-F}.$$

The Prandtl number is

$$\begin{aligned} Pr &= C_p \, \mu/k \\ &= (1.258 \text{ Btu/lbm-F})(0.2336 \text{ lbm/hr-ft})/0.3346 \text{ Btu/hr-ft-F}) \\ &= 0.878. \end{aligned}$$

From the circular tube Nusselt number correlation,

$$\begin{aligned} Nu_{circ} &= 0.023 \, Re^{0.8} \, Pr^{0.4} \\ &= 0.023 \, (344{,}600)^{0.8} \, (0.878)^{0.4} = 587.5 . \end{aligned}$$

Applying the correction for the actual flow channel,

$$Nu = Nu_{circ}\ [-3.475 + 8.053(S/D_s) - 4.705(S/D_s)^2 + 0.916(S/D_s)^3]$$

$$= (587.5)\left[-3.475 + 8.053\left(\frac{0.45}{0.35}\right) - 4.705\left(\frac{0.45}{0.35}\right)^2 + 0.916\left(\frac{0.45}{0.35}\right)^3\right]$$

$$= 615.7\ .$$

Calculation of h from the definition of Nu gives

$$h = (Nu)\ k/D_e$$

$$= (615.7)(0.3346\ \text{Btu/hr-ft-F})/(0.0322\ \text{ft})$$

$$= 6397\ \text{Btu/hr-ft-F}\ .$$

4.7 Incremental Energy Balances

As heat is transferred from the fuel rod to the coolant, the temperature of the coolant rises. Thus, the coolant temperature varies from a minimum at the inlet of the coolant channel to a maximum at the exit of the fuel bundle. The bulk temperature at any axial position can be determined from an energy balance where the total energy added by heat transfer to the coolant is equated to the energy rise of the coolant, i.e.,

$$\dot{m}i = \dot{m}i_{in} + \int_{-H/2}^{Z} q_s\,(Z)\,dZ. \tag{57}$$

Recognizing that for subcooled water $\Delta i = C_p \Delta T_B$ and solving for the local bulk temperature, this becomes

$$T_B = T_{in} + \frac{1}{\dot{m}\,C_p} \int_{-H/2}^{Z} q_s\,(Z)\,dZ. \tag{58}$$

Equation 58 can be used to determine the bulk temperature increase for any segment, ΔL, of the coolant channel by

$$T_{B2} - T_{B1} = \frac{1}{\dot{m}\;C_p} \int_{Z_c-\Delta L/2}^{Z_c+\Delta L/2} q_s(Z)\,dZ \tag{59}$$

where Z_c is the center of the increment under consideration. Defining an average surface heat flux for the increment, $\bar{q}_s'' \equiv \bar{q}_s/A_s$, the incremental bulk temperature rise becomes

$$T_{B2} - T_{B1} = \frac{\pi D_s \Delta L}{\dot{m}\,C_p}\;\bar{q}_s''. \tag{60}$$

The local surface heat flux, $q_s''(Z)$, is related to the local volumetric thermal source strength by observing that all of the energy generated in the fuel is transferred out through the surface of the cladding; i.e.,

$$\pi D_s \, dZ \, q_s''(Z) = \frac{\pi}{4} D_f^2 \, dZ \, q'''(Z). \tag{61}$$

This simplifies to

$$q_s''(Z) = \frac{D_f^2}{4D_s} q'''(Z). \tag{62}$$

Similarly,

$$\bar{q}_s'' = \frac{D_f^2}{4D_s} \bar{q}'''. \tag{63}$$

4.8 Assessment of Local Coolant Conditions

Two important checks on coolant conditions should be made to insure that the reactor is adequately cooled. The first is a determination of whether the fluid bulk is indeed subcooled. As the coolant flows through the reactor, its energy is increased by heat transfer and its pressure is decreased by flow losses. Both of these effects reduce the subcooling of the fluid. To insure bulk subcooling, the local bulk temperature must be less than the saturation temperature at the local pressure. The critical location, of course, is at the exit of the channel, where the coolant bulk temperature is the maximum and the pressure is at a minimum.

The other determination that should be made is whether there is any local boiling near the surface of the fuel rod. The criterion for this determination is whether the maximum coolant temperature at any cross section is greater than the local saturation temperature. The maximum coolant temperature at any cross section occurs in the fluid layer immediately adjacent to the cladding and its magnitude approaches the surface temperature of the cladding. The axial location of the maximum cladding surface temperature can be obtained by writing an expression for $T_s(Z)$ in terms of q'''_o, H_e, D_f, D_s, $\dot{m}$, C_p and h, differentiating with respect to Z, setting the derivative equal to zero, and solving for Z_m. The maximum cladding surface temperature can then be obtained by evaluating $T_s(Z)$ at Z_m. The details of this calculation are left as an exercise for the student.

4.9 Subcooled Flow Boiling

Even though the bulk fluid is subcooled for PWR operation the possibility of local or subcooled flow boiling exists. Whenever the outer cladding temperature is greater than the local saturation temperature subcooled boiling may occur. Such conditions are obtained in PWR'S during peak loads. Under these conditions there is a possibility of boiling burnout of the surface. To assess this possibility, the actual heat flux at the surface, q''_s, is compared to the critical heat flux, q''_c, computed for subcooled flow boiling. The critical heat flux (CHF), also called the burnout heat flux or departure from nucleate boiling (DNB), corresponds to the heat flux at which blanketing of the heat transfer surface by coalesced vapor bubbles is just imminent. Any further increase in heat flux would require the heat transfer to occur through the vapor blanket. Because of the low conductivity of the vapor blanket, a catastrophic increase in surface temperature would result and generally lead to melting of the cladding.

One of the most widely used subcooled flow boiling critical heat flux correlations is the ANL correlation of Jens and Lottes [13]. This correlation is based on a large number of experimental data for subcooled water flowing vertically upward in a flow channel under a range of operating conditions. The conditions spanned by the data are mass velocity from 0.96 x 10^6 to 7.8 x 10^6 lbm/hr-ft, pressure from 500 to 3000 psia, and subcooling from 5 to 163 F. The correlation for the critical heat flux,

$$q''_c = C \left(\frac{G}{10^6}\right)^m (T_{sat} - T_B)^{0.22}, \tag{64}$$

contains two pressure dependent "constants" which are given in Table 1 below.

TABLE 1

The "constants" C and m for Jens and Lottes Correlation

Water Pressure, psia	C	m
500	0.817×10^6	0.160
1000	0.626×10^6	0.275
2000	0.445×10^6	0.500
3000	0.250×10^6	0.730

Note that the Jens and Lottes correlation breaks down as the subcooling approaches zero. For this case the more complex Westinghouse correlation [14], valid for high subcooling to 15% quality, is recommended.

Whenever subcooled flow boiling exists to any appreciable extent, the cladding surface temperature can no longer be calculated from Newton's law of cooling. This is because the heat transfer coefficient in Newton's law of cooling does not reflect the increased agitation in the convective (now boiling) film due to the growth of the vapor bubbles. Instead, the correlation of Jens and Lottes [13] for subcooled boiling,

$$\Delta T_{sat} = 1.9\,(q_s'')^{0.25}\, e^{-P/900}, \qquad (65)$$

may be used. In this relation, ΔT_{sat} is the difference between the surface temperature and the local saturation temperature.

5.0 The PWR Thermal-Hydraulics Code

The PWR Thermal-Hydraulics Code calculates the thermal-hydraulics performance parameters discussed in Section 4 for the PWR model described in Section 3. The basic operational steps of the code are listed below.

1. Real↔integer conversions
2. Definition of statement functions
3. Accept and print input
4. Calculate flow area and equivalent diameter of coolant channel
5. Determine calculation increment
6. Calculate inlet pressure loss
7. Initialize to 1st calculation increment
8. Calculate bulk temperature rise and average temperature for 1st increment
9. Calculate average coolant properties for 1st increment
10. Calculate pressure loss for 1st increment
11. Calculate average pressure and saturation temperature for 1st increment
12. Calculate heat transfer coefficient for 1st increment
13. Calculate cladding outer surface temperature for 1st increment
14. Check for bulk boiling, surface boiling, critical heat flux for 1st increment
15. Calculate temperature distribution in fuel for 1st increment
16. Print 1st increment output
17. Repeat steps 7-16 for remaining increments
18. Calculate exit pressure loss and total pressure loss
19. Print coolant exit conditions.

The details of these specific steps requiring further explanation are found in the discussion below. A flow chart showing the calculations and logic is given in Section 5.4.

Most of the statement functions are used for convenience in calculating property data. These statement functions are in the form of some property, either of the coolant, cladding, or fuel, as a function of the corresponding temperature. All of the function forms are polynomials and were obtained by fitting polynomials of various degree to tabulated data. The order of these polynomials varies, having been selected to insure that it is accurate to within 2% over the range of interest of the parameter. Similar polynomial fit statement functions are used to relate the local saturation temperature and the "constants" C and m in Equation 64 to the local coolant pressure.

A detailed discussion of the input requirements is deferred until Section 5.2. For the moment it suffices to note that this input is supplied on three cards; one each for the geometry, the inlet flow conditions, and the reactor power level.

Since the cross section of the model flow channel does not change with axial position, the flow area and equivalent diameter are constant axially. The flow area is calculated from

$$A_f = S^2 - \frac{\pi}{4} D_f^2 \qquad (66)$$

and the equivalent diameter from

$$D_e = \frac{4A_f}{P_w} \qquad (30)$$

where the wetted perimeter is given by

$$P_w = \pi D_f. \qquad (67)$$

In the code, the model coolant channel and fuel rod are sliced into short segments which are stacked axially to form the proper length core. The actual length of these segments is selected as a compromise between the very short segments for which the assumption of constant properties over the section axially is accurate and the very long segment which minimizes the computer time. An increment size equal to 1/100 of the active core length is built into the code. To study the effect of increment size, or to save computer time, this can readily be changed by altering two lines of the code.

After the increment size has been computed, the coolant conditions at the beginning of the first increment are established. The bulk temperature and mass velocity at the entrance of the first increment are set equal to the core inlet conditions provided in the input. The pressure at the entrance of the first calculation increment is obtained by subtracting the inlet pressure loss, computed from Equation 26 and Equation 34, from the pressure at the inlet of the core. In addition, counters required in the code logic are set equal to 1, denoting the first increment.

The first major calculation in the code is the determination of the bulk temperature rise for the first calculation increment. This is calculated from Equation 60 with $\overline{q}_s''$ determined from Equation 63. For small calculation increments, the average volumetric thermal source strength can be accurately approximated by evaluating Equation 1 at $-H/2 + \Delta L/2$; i.e., at the center of the increment.

The average bulk temperature, $\bar{T}_B$, for the increment is taken as the arithemetic average of the inlet and exit values. All coolant properties used in the pressure drop, heat transfer coefficient and critical heat flux calculations are assumed to be functions of temperature only and are evaluated at the average bulk temperature.

Both the friction factor and heat transfer coefficient calculations require the Reynolds number of the flow which is calculated from Equation 31. Then f_{circ} and Nu_{circ} are calculated from Equations 32 and 55, respectively. To these values, the multiplicative correction factors are applied to yield f and Nu for the actual reactor flow. Knowing f, the pressure drop for the increment is calculated from Equation 26. The heat transfer coefficient is backed out of the Nu definition, Equation 52.

The average outer surface temperature of the cladding for the increment is determined from Newton's law of cooling with $\bar{q}_s''$, $\bar{T}_B$ and h available from previous calculations for the increment; i.e.,

$$\bar{T}_s = \bar{T}_B + \frac{\bar{q}_s''}{h} . \tag{68}$$

At this point, several comparisons are built into the code which check for proper reactor operation. The first comparison is of the average bulk temperature to the saturation temperature evaluated at the average pressure of the increment. $\bar{T}_B < \bar{T}_{sat}$ is required for subcooled operation. If improper operation is observed an informative message is printed and the calculations are terminated. The second comparison is $\bar{T}_s$ to $\bar{T}_{sat}$. If $\bar{T}_s > \bar{T}_{sat}$, then surface boiling may be present. The possibility of surface boiling for the increment is indicated by a printed message. If surface boiling is indicated by the above criterion, then a comparison of $\bar{q}_s''$ to the critical heat flux calculated from Equation 64 is made. The ratio $\bar{q}_s''/\bar{q}_c'' \equiv$ CHFR (critical heat flux ratio) is computed for all segments in which surface boiling is indicated. If CHFR is equal to one or greater, the program terminates after printing out $\bar{q}_s''$ and $\bar{q}_c''$. If surface boiling exists but burnout does not occur, the cladding surface temperature is recalculated from Equation 65.

After the coolant channel calculations for the increment have been performed, the radial temperature distribution in the fuel is calculated for the increment.

First, the temperature at the outer surface of the fuel is found. In the fuel rod model, the outer surface of the fuel is at the same temperature as the inner surface of the cladding; therefore, the outer surface fuel temperature, $\bar{T}_f$, can be obtained by adding the temperature drop across the cladding to the outer surface temperature of the cladding, $\bar{T}_s$, which was calculated earlier.

The temperature drop across the cladding is obtained from Equation 22. In Equation 22 the thermal conductivity of the cladding appears. This property is temperature dependent and should be evaluated at the mean temperature of the cladding. However the mean cladding temperature is dependent on the temperature rise across the cladding. Thus, an iterative-type solution is called for. The procedure used is to evaluate k_{c-I} at $\bar{T}_s$, calculate $\bar{T}_f$, calculate $\bar{T}_{clad\ mean}$, evaluate k_{c-II} at $\bar{T}_{clad\ mean}$, compare k_{c-II} to k_{c-I}. If k_{c-II} is within 1.0% of k_{c-I}, accept value of $\bar{T}_f$. If k_{c-II} is not within 1.0% of k_{c-I}, let $k_{c-I} = k_{c-II}$ and repeat the sequence. In this way a cladding thermal conductivity accurate to within 1.0% is used.

The next step is to calculate the temperature at the center of the fuel from Equation 14. Again a temperature dependent thermal conductivity, k_f, appears, and an iterative calculation procedure like that described in the previous paragraph is used. The fuel temperature at the quarter, half, and three-quarter radii are then calculated from Equation 12.

After the calculations for the first increment are completed the results are printed out. Discussion of the output is deferred until Section 5.3. It should be noted here, however, that in the output tabulation the temperatures are reported as having occurred at the axial center of the increment. In other words, for the first increment the results are reported at $Z = - H/2 + L/2$. After the results for the first increment have been printed out, the code increments to the next segment, and continues this process until all 100 segments have been spanned.

After the completion of the calculations and printout for the 100 segments of the core, the core exit conditions are calculated and printed. A calculation of the exit pressure loss using Equations 26 and 34 is required. Then P_{ex} and T_{ex} are computed and printed, ending the computations.

5.1 Nomenclature

Analysis Symbol	Code Symbol	Description
A		Area
A_f	AFL	Flow area of coolant channel
A_n		Heat transfer area normal to n-direction
A_r		Heat transfer area at radius r
A_s		Heat transfer area at outside of cladding
A_w		Surface area of wall
C	C	Constant in Equation 64
C_1, C_2, C_3, C_4		Integration constants
C_p	CP	Constant pressure specific heat
CHFR	CHFR	Critical heat flux ratio
D		Diameter
D_e	DEQ	Equivalent diameter of coolant channel
D_f	DFU	Outside diameter of nuclear fuel
D_s	DCL	Outside diameter of cladding
E_f		Energy release per fission
e	ERR	Error ratio
f	F	Friction factor
f_{circ}	FCIRC	Friction factor for circular tubes
f/f_{circ}	FRAT	Ratio of actual to circular tube friction factor
g_c		Dimensional constant = 32.174 $\frac{lbm\text{-}ft}{sec^2\text{-}lbf}$

Analysis Symbol	Code Symbol	Description
G	GIN	Mass velocity of coolant
h	HTC	Convective heat transfer coefficient
H	H	Height of active core
H_e	HE	Extrapolated height of core
i		Enthalpy
i_{in}		Enthalpy at inlet of core
k		Thermal conductivity
k_B	KBC	Thermal conductivity of the coolant
k_c	KCL	Thermal conductivity of cladding
k_{c-I}	KCL1	Thermal conductivity of cladding - at pre-iteration average cladding temperature
k_{c-II}	KCL2	Thermal conductivity of cladding - at updated average cladding temperature
k_f	KFU	Thermal conductivity of fuel
	KFU1	Thermal conductivity of fuel - at pre-iteration average fuel temperature
	KFU2	Thermal conductivity of fuel - at updated average fuel temperature
K		Flow resistance coefficient
L_e		Extrapolation length
ΔL	DELL	Length increment
m	M	Exponent in Equation 64
$\dot{m}$	MDØT	Mass flow rate
n		Exponent in Equation 55
N_{ff}		Number of fissionable fuel nuclei per unit volume

Analysis Symbol	Code Symbol	Description
Nu	NU	Nusselt number
Nu_{circ}	NUCIRC	Nusselt number for circular tubes
Option	ØP	Option = 1 for cosine, Option = 2 for coupled program
P_w	PW	Wetted perimeter
P	PAV	Pressure
P_1	P1	Pressure at beginning of increment
P_2	P2	Pressure at end of increment
P_{ex}	PEX	Core exit pressure
P_{in}	PIN	Core inlet pressure
Pr	PR	Prandtl number
Pr_t		Turbulent Prandtl number
ΔP		Pressure drop
ΔP_E	DELPE	Exit or entrance pressure loss
ΔP_F	DELPF	Frictional pressure loss
q		Heat transfer rate
q_f		Heat transfer rate at D_f
q_n		Heat transfer rate in n-direction
q_s		Heat transfer rate at D_s
q_w		Heat transfer rate at wall
q''_c	CHF	Critical heat flux
q''_s	SHFAV	Heat flux at D_s
q'''	QTPAV	Volumetric thermal source strength
q'''_o	QTPO	Volumetric thermal source strength at center of core
r		Fuel pin radial coordinate
r_f		Radius of fuel

Analysis Symbol	Code Symbol	Description
r_s		Outer radius of cladding
R		Core radial coordinate
Re	RE	Reynolds number
S	S	Center-to-center spacing between fuel rods
S/D_s	SRAT	Ratio of fuel rod spacing to fuel rod diameter
St		Stanton number
T	T	Temperature
T_o	TO	Temperature at center of fuel rod
$T_{1/4}$	T14	Temperature in fuel at $r = 1/4\ r_f$
$T_{1/2}$	T12	Temperature in fuel at $r = 1/2\ r_f$
$T_{3/4}$	T34	Temperature in fuel at $r = 3/4\ r_f$
T_B	TBAV	Coolant bulk temperature
T_{B1}	TB1	Bulk temperature at beginning of increment
T_{B2}	TB2	Bulk temperature at end of increment
T_f	TF	Temperature in fuel at D_f
T_{ex}	TEX	Core exit temperature
T_{in}	TIN	Core inlet temperature
T_s	TSAV	Cladding surface temperature
T_{sat}	TSATAV	Saturation temperature
T_w		Wall temperature
U		Velocity
U_B	UB	Average velocity in flow channel
X_n		Coordinate in n-direction
y		Coordinate normal to wall
z		Axial coordinate

Analysis Symbol	Code Symbol	Description
Z_c	ZC	Axial location of center of increment
Z_m		Axial location of maximum cladding surface temperature
α		Thermal diffusivity
ε		Roughness size
ε_H		Eddy diffusivity of heat
ε_M		Eddy diffusivity of momentum
θ		Angular coordinate
μ	VIS	Absolute viscosity of water
ν		Kinematic viscosity of water
ρ	RHØ	Density of water
$\bar{\sigma}$		Average fission microscopic cross section
τ		Shear stress; also time
τ_w		Shear stress at wall
ϕ		Neutron flux
superscript --		Average for calculation increment

5.2 Code Input

The input to the PWR Thermal-Hydraulics Code is intentionally extremely simple. Only three cards are required; the first lists the geometrical quantities, the second lists the coolant inlet conditions, and the third lists the parameters that describe the volumetric thermal source strength distribution in the reactor.

The required input data cards are listed below. The units of each parameter and their input format are included. Sample input data cards are shown in Figure 2.

Card 1 -- Geometry

D_f	(in.)	F 10.4
D_s	(in.)	F 10.4
S	(in.)	F 10.4
H	(in.)	F 10.4

Card 2 -- Coolant inlet conditions

G	($lbm/hr-ft^2$)	E 10.3
T_{in}	(F)	F 10.1
P_{in}	(psia)	F 10.1

Card 3 -- Power distribution

q'''_o	($Btu/hr-ft^3$)	E 10.3
H_e	(in.)	F 10.1
Option	(pure number)	I 10

On Card 3 the axial volumetric thermal source strength distribution through the core is specified. For independent operation of the code

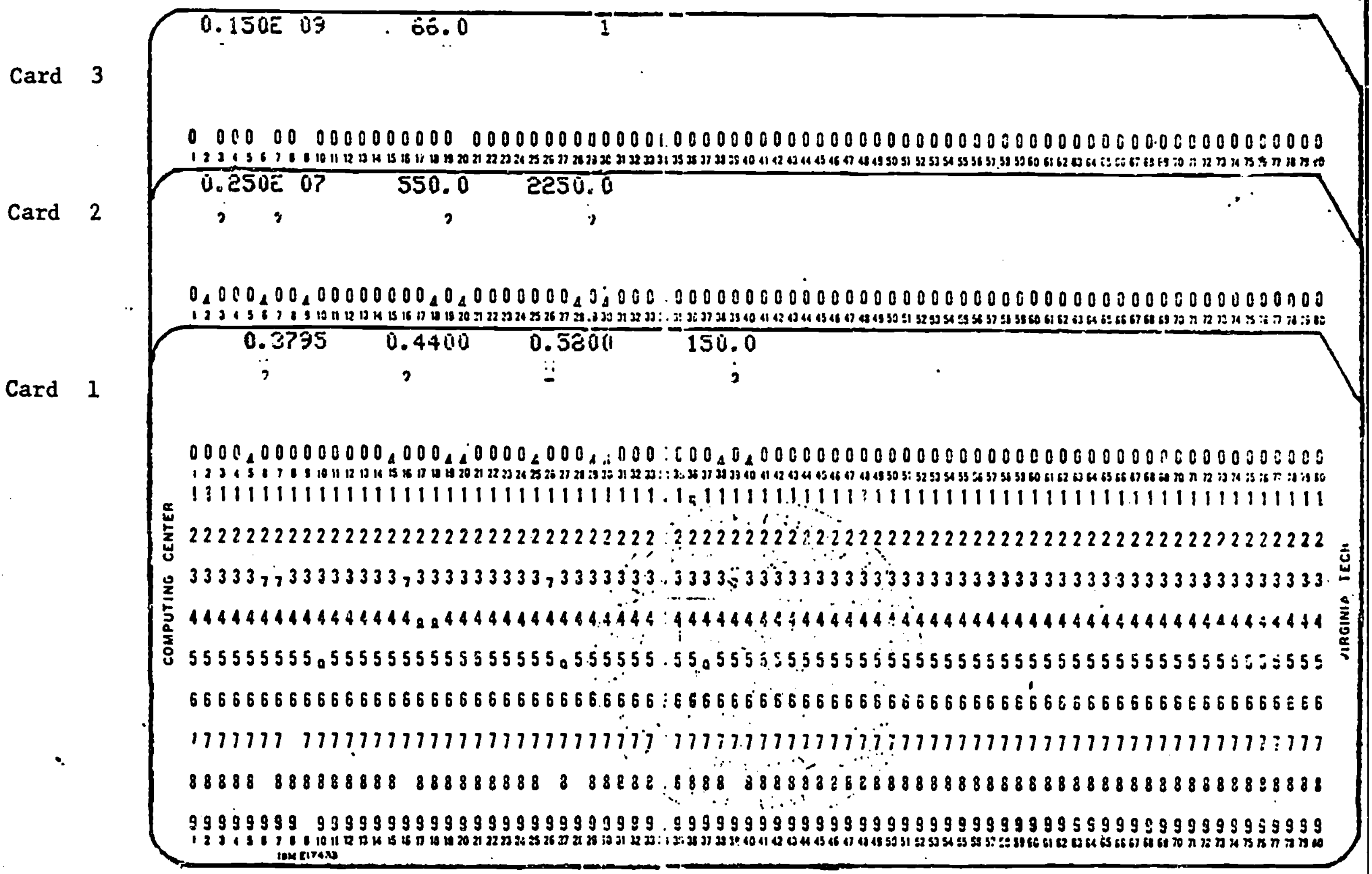


Figure 2. PWR Thermal-Hydraulics Code Input.

(Option = 1) a cosine distribution is built into the code and q'''_o and H_e are the parameters which indicate its level and period. When the PWR Thermal-Hydraulics Code is coupled with other codes (Option = 2) the cosine distribution is over-ridden by the actual power distribution which is supplied by the main program.

5.3 Code Output

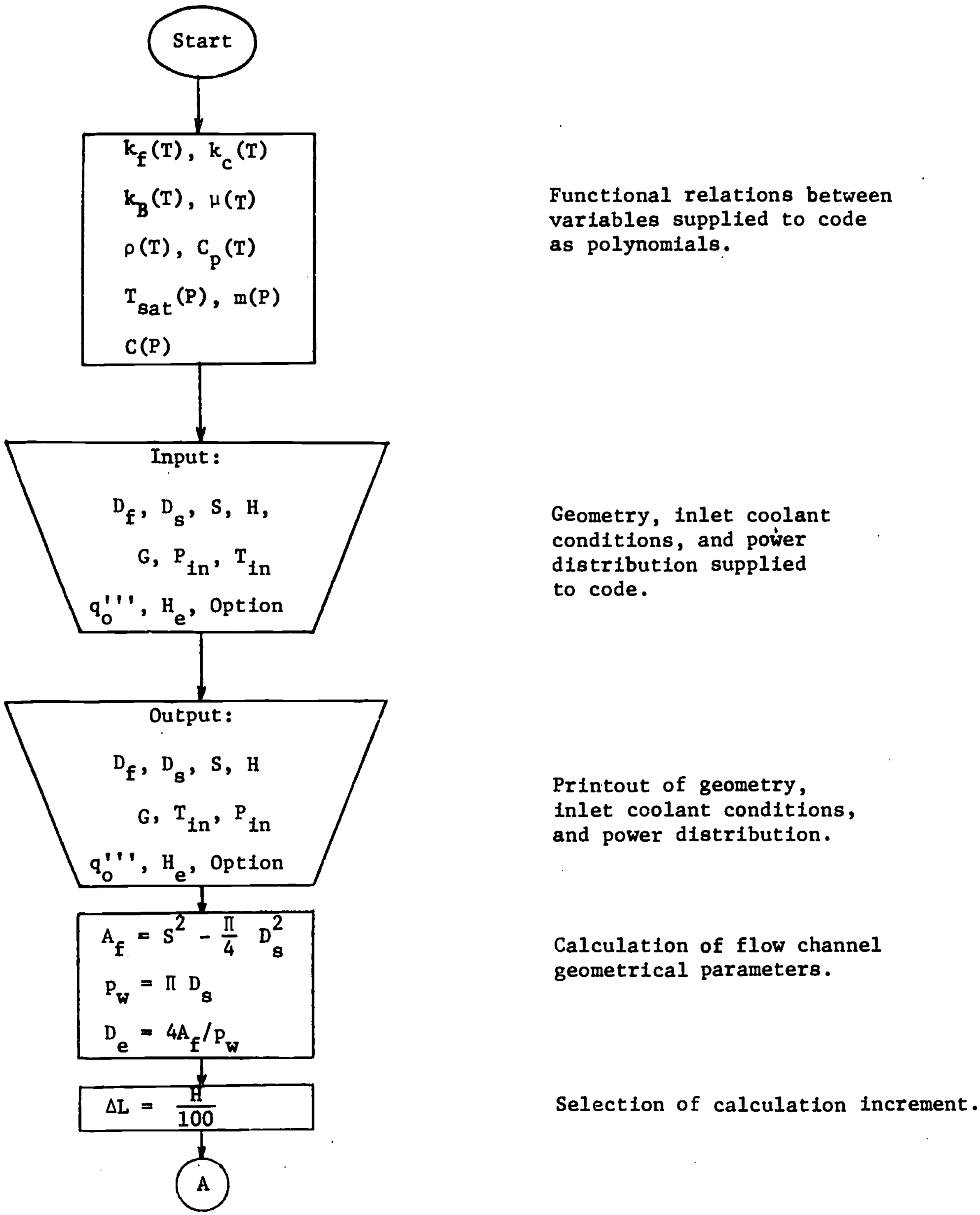
Three sets of information are printed out. Each is printed under completely explicit headings such that there should be no interpretation difficulties.

The first set of output information consists of a listing of the input that was supplied to the code.

The second and main set of output is a tabulation of the temperatures, h, ΔP, and CHFR for each increment in the core. The temperatures printed out are $\bar{T}_o$, $\bar{T}_{1/4}$, $\bar{T}_{1/2}$, $\bar{T}_{3/4}$, $\bar{T}_f$, $\bar{T}_s$, $\bar{T}_B$, $\bar{T}_{sat}$. From this data, observations of the maximum fuel and cladding temperatures and their locations can be made.

The third set of data reports the core exit pressure and temperature.

5.4 Code Flow Chart



A

$$U_B = G/\rho(T_{in})$$

$$Re = \frac{U_B D_e \, \rho(T_{in})}{\mu(T_{in})}$$

$$f_{circ} = 0.184/Re^{0.2}$$

$$f = f_{circ} \, \frac{f}{f_{circ}} \left(\frac{S}{D_s}\right)$$

$$\Delta P_F = f \, \frac{\Delta L}{D_e} \, \frac{\rho(T_{in}) U_B^2}{2g_c}$$

$$\Delta P_E = K \, \frac{\rho(T_{in}) U_B^2}{2g_c}, \quad K = 0.5$$

$$P_1 = P_{in} - \Delta P_E - \Delta P_F$$

Calculation of permanent pressure loss at inlet and resulting pressure at start of active core.

$$T_{B1} = T_{in}$$

$$Z_c = - H/2 + \Delta L/2$$

$$I = 1$$

Initializing temperature, position, and increment counter for 1st increment.

B

$$\bar{q}''' = q_o''' \cos \frac{\Pi Z_c}{H_e}$$

$$T_{B2} = T_{B1} + \bar{q}''' \, \Pi D_f^2/4\dot{m}C_p(T_{B1})$$

$$\bar{T}_B = (T_{B1} + T_{B2})/2$$

Calculation of average bulk temperature of coolant for increment.

C

C

$$U_B = G/\rho(\bar{T}_B)$$

$$Re = \frac{U_B D_e \, \rho(\bar{T}_B)}{\mu(\bar{T}_B)}$$

$$f_{circ} = 0.184/Re^{0.2}$$

$$f = f_{circ} \, \frac{f}{f_{circ}}\left(\frac{S}{D_s}\right)$$

$$\Delta P_F = f \, \frac{\Delta L}{D_e} \, \frac{\rho(\bar{T}_B) U_B^2}{2 g_c}$$

$$P_2 = P_1 - \Delta P_F$$

$$\bar{P} = (P_1 + P_2)/2$$

$$\bar{T}_{sat} = T_{sat}\,(\bar{P})$$

Calculation of pressure drop and average saturation temperature for increment.

$$Pr = \frac{C_p(\bar{T}_B)\, \mu(\bar{T}_B)}{k_B(\bar{T}_B)}$$

$$Nu_{circ} = 0.023 \; Pr^{0.4} \; Re^{0.8}$$

$$Nu = Nu_{circ} \, \frac{Nu}{Nu_{circ}}\left(\frac{S}{D_s}\right)$$

$$h = \frac{Nu \; k_B \; (\bar{T}_B)}{D_e}$$

$$\bar{T}_s = \bar{T}_B + \frac{\bar{q}''' D_f^2}{4 \, D_s \, h}$$

Calculation of heat transfer coefficient and average cladding surface temperature for increment.

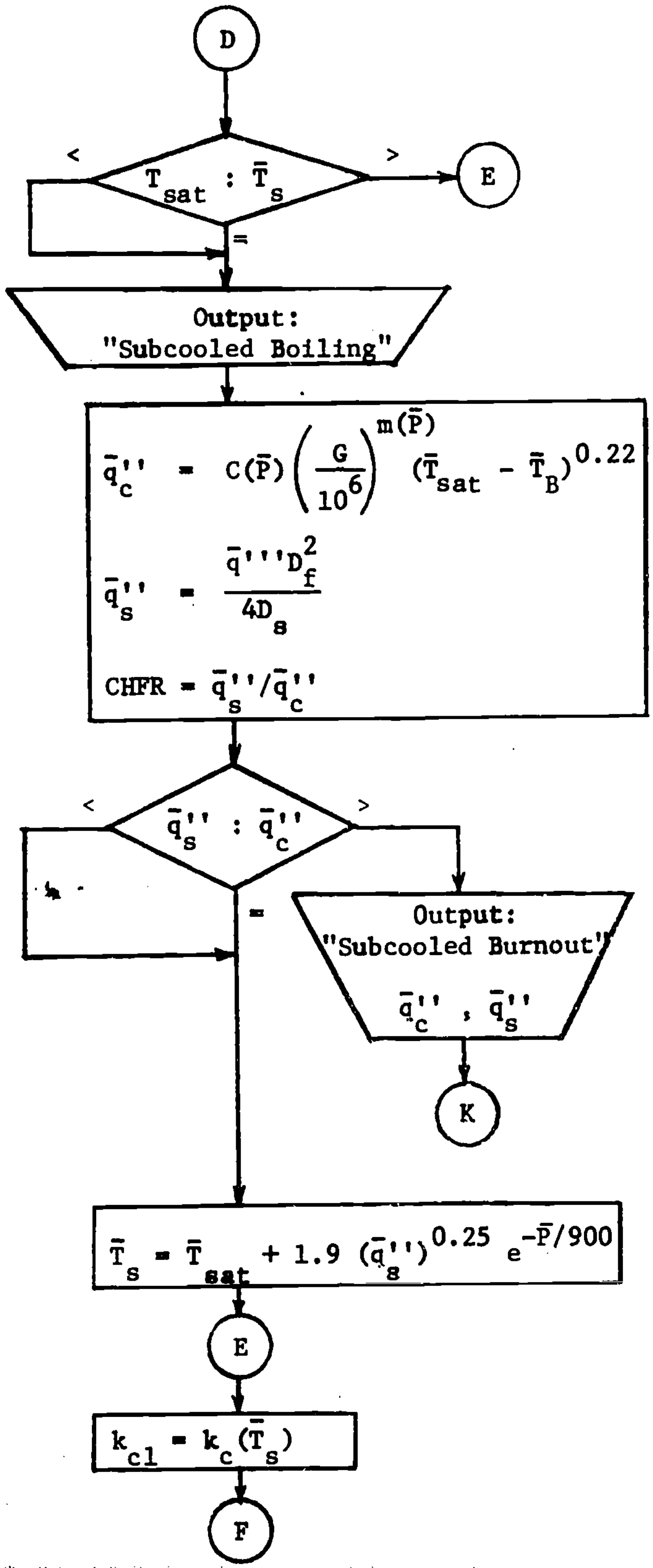
$T_{B2} : \bar{T}_{sat}$

$<$: D

$=$: D

$>$: Output: "Net Void Formation" → K

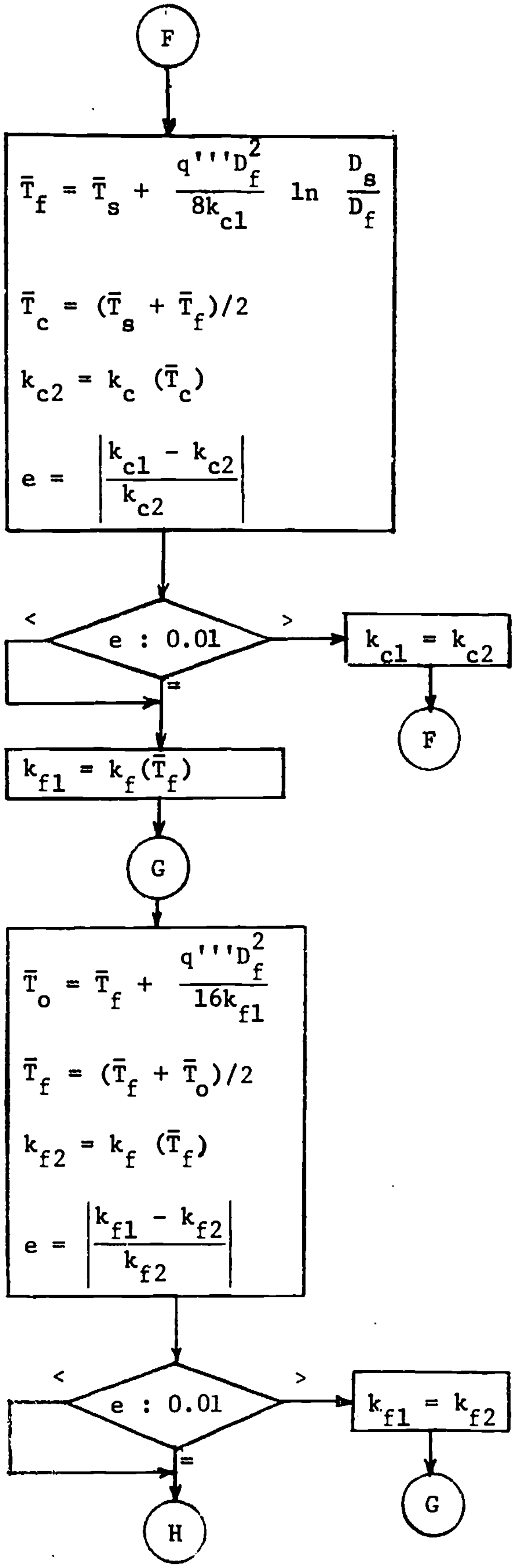
Check to see if bulk coolant remains subcooled. Error message printed and calculations terminated if bulk coolant reaches saturation.



Check to see if subcooled boiling is possible. If possibility is detected, a message is printed and check for subcooled boiling burnout is initiated. If no boiling is indicated, program proceeds to calculation of fuel and cladding temperature distributions.

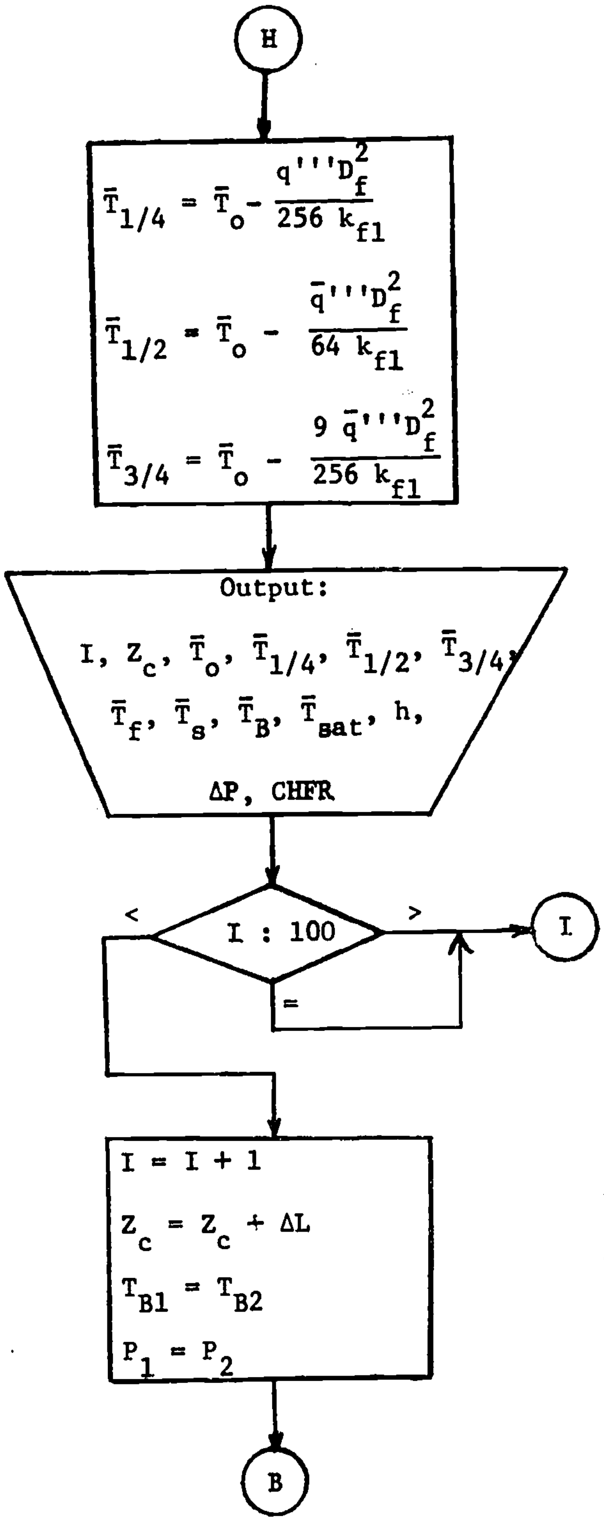
Critical heat flux check. If positive, message and local conditions printed out and calculations terminated.

Recalculation of cladding surface temperature for subcooled boiling on surface.



Calculation of temperature rise across cladding. The thermal conductivity of the cladding is evaluated at the mean cladding temperature by an iterative technique.

Calculation of fuel centerline temperature. The thermal conductivity of the fuel is evaluated at a mean fuel temperature by an iterative technique.

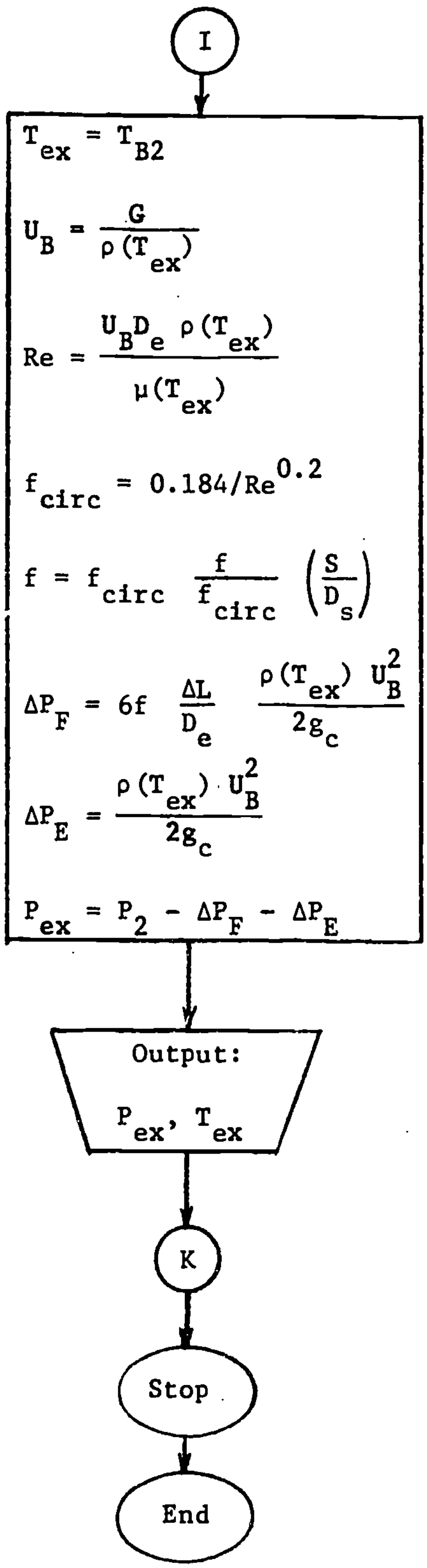


Evaluation of fuel temperature at the quarter, half, and three quarter diameter points.

Printout of thermal-hydraulic parameters for increment.

Check to see if entire core has been calculated

Initializing for next calculation increment.



Calculation of pressure loss in gas plenum and downstream rod support region.

Printout of coolant core exit conditions.

5.5 Code Examples

The following two examples demonstrate the use of the PWR Thermal-Hydraulics Code for calculating the thermal-hydraulic behavior of a typical PWR.

Example 7.

A PWR core is composed of 0.422 in. diameter fuel rods 12 ft long and spaced 0.563 in. in a square array. The fuel diameter is 0.3664 in. The inlet conditions are 543 F and 2250 psia. The average coolant velocity in the inlet of the core is 13.02 ft/sec. The average fuel rod produces 5.95 kw/ft. Assume an extrapolation length equal to 5% of the core height. Determine the magnitudes and locations of the maximum fuel temperature and the maximum cladding temperature.

Solution.

The mass velocity, extrapolated core height, and peak volumetric thermal source strength must first be computed from the data given.

$$G = \rho U_B = (47.0 \text{ lbm/ft}^3)(13.02 \text{ ft/sec})(3600 \text{ sec/hr})$$

$$= 2.2 \times 10^6 \text{ lbm/hr-ft}^2$$

$$H_e = H + 2L_e = H + 2(0.05H) = 1.10H$$

$$= 1.10 \text{ (12 ft)} = 13.2 \text{ ft} = 158.4 \text{ in.}$$

$$q_o''' = 2P_{ave}/[H_e\ D_f^2 \sin(\Pi H/Z\ H_e)]$$

$$= 2(5.95 \text{ kw/ft})(12 \text{ ft})(3413 \text{ Btu/kw-hr})/\{(13.2 \text{ ft})(0.3664 \text{ in.})^2$$

$$(\text{ft}/12 \text{ in.})^2 \sin[\Pi(12 \text{ ft})/2(13.2 \text{ ft})]\}$$

$= 4.0 \times 10^7$ Btu/hr-ft^3.

From the output printout of the code (see next five pages) we find

$$T_{clad\ max} = 723.3 \text{ F at } Z = 0.960 \text{ ft}$$

$$T_{fuel\ max} = 1677.6 \text{ F at } Z = 0.180 \text{ ft.}$$

```
******************************
 PWR THERMAL HYDRAULICS CODE
******************************

 INPUT DATA

 FUEL O.D.        CLAD O.D.        ROD PITCH        ACTIVE CORE LENGTH
 ( INCH )         ( INCH )         ( INCH )             ( INCH )
  0.3664           0.4220           0.5630               144.00

 INLET MASS VELOCITY   CORE INLET TEMPERATURE   CORE INLET PRESSURE
   ( LB/HR-FT**2 )             ( F )              (    PSIA    )
      0.2200E 07              543.00                 2250.00

 VOLUMETRIC THERMAL    EXTRAPOLATED HEIGHT
  ( BTU/HR-FT**3 )           ( INCH )
     0.4000E 08               158.40

                    OPTION    1

 E N D  O F  I N P U T  D A T A
 ******************************



   AT THE INLET OF THE CORE

CHANNEL AVE. VELOCITY   REYNOLDS #   FRICTION FACTOR
      (FT/SEC)
       13.02            413202.1         0.0139

**************************************************************************
```

NO	Z-LOC.	TO	T1/4	T1/2	T3/4	TF	T CLAD	T BULK	T SAT	H T COEF	DELTA P	CHF RATIO
1	-5.940	663.3	660.3	639.0	608.7	566.1	550.6	543.1	652.6	5497.1	4.94	0.0
2	-5.820	686.3	682.7	657.4	621.2	570.5	552.2	543.3	652.6	5497.9	4.94	0.0
3	-5.700	709.6	705.4	675.9	633.8	574.8	553.7	543.5	652.6	5498.8	4.94	0.0
4	-5.580	733.2	728.4	694.7	646.5	579.1	555.3	543.7	652.6	5499.9	4.94	0.0
5	-5.460	757.0	751.6	713.6	659.4	583.4	556.9	544.0	652.6	5501.0	4.94	0.0
6	-5.340	781.1	775.1	732.8	672.3	587.7	558.5	544.3	652.6	5502.3	4.94	0.0
7	-5.220	806.8	800.1	753.1	686.0	592.0	560.1	544.6	652.6	5503.7	4.94	0.0
8	-5.100	831.7	824.4	772.8	699.3	596.3	561.8	544.9	652.6	5505.3	4.95	0.0
9	-4.980	856.8	848.8	792.8	712.7	600.5	563.4	545.3	652.6	5507.0	4.95	0.0
10	-4.860	882.2	873.6	812.8	726.1	604.8	565.1	545.7	652.6	5508.7	4.95	0.0
11	-4.740	907.8	898.5	833.1	739.7	609.0	566.8	546.2	652.6	5510.6	4.95	0.0
12	-4.620	933.5	923.5	853.4	753.3	613.2	568.5	546.6	652.6	5512.7	4.95	0.0
13	-4.500	959.4	948.8	873.9	767.0	617.3	570.1	547.1	652.6	5514.8	4.96	0.0
14	-4.380	985.5	974.1	894.5	780.7	621.5	571.8	547.6	652.6	5517.0	4.96	0.0
15	-4.260	1011.6	999.6	915.1	794.4	625.5	573.5	548.2	652.6	5519.4	4.96	0.0
16	-4.140	1037.8	1025.1	935.8	808.2	629.6	575.2	548.7	652.6	5521.8	4.96	0.0
17	-4.020	1064.1	1050.7	956.4	821.9	633.6	577.0	549.3	652.6	5524.4	4.97	0.0
18	-3.900	1089.5	1075.4	976.3	834.9	636.9	578.7	549.9	652.6	5527.0	4.97	0.0
19	-3.780	1115.6	1100.9	996.9	848.5	640.7	580.4	550.5	652.6	5529.8	4.97	0.0
20	-3.660	1141.7	1126.2	1017.4	862.0	644.5	582.1	551.2	652.6	5532.6	4.98	0.0
21	-3.540	1167.6	1151.5	1037.8	875.5	648.3	583.7	551.9	652.6	5535.5	4.98	0.0
22	-3.420	1193.4	1176.6	1058.0	888.8	651.9	585.4	552.6	652.6	5538.5	4.99	0.0
23	-3.300	1223.0	1205.4	1081.1	903.8	655.5	587.1	553.3	652.5	5541.6	4.99	0.0

24	-3.180	1248.9	1230.5	1101.4	917.1	659.0	588.8	554.0	652.5	5544.8	4.99	0.0
25	-3.060	1274.5	1255.4	1121.5	930.2	662.5	590.4	554.8	652.5	5548.1	5.00	0.0
26	-2.940	1299.7	1280.0	1141.3	943.2	665.9	592.1	555.5	652.5	5551.4	5.00	0.0
27	-2.820	1324.6	1304.2	1160.8	955.9	669.2	593.7	556.3	652.5	5554.8	5.01	0.0
28	-2.700	1349.1	1328.0	1179.9	968.5	672.4	595.3	557.1	652.5	5558.2	5.01	0.0
29	-2.580	1373.1	1351.4	1198.7	980.7	675.6	596.9	557.9	652.5	5561.7	5.01	0.0
30	-2.460	1396.5	1374.2	1217.1	992.7	678.6	598.5	558.8	652.5	5565.3	5.02	0.0
31	-2.340	1419.4	1396.4	1234.9	1004.4	681.6	600.0	559.6	652.5	5568.9	5.02	0.0
32	-2.220	1441.6	1418.0	1252.3	1015.7	684.5	601.6	560.5	652.5	5572.5	5.03	0.0
33	-2.100	1463.1	1438.9	1269.1	1026.7	687.3	603.1	561.4	652.5	5576.2	5.03	0.0
34	-1.980	1483.8	1459.1	1285.4	1037.3	690.0	604.6	562.3	652.5	5579.9	5.04	0.0
35	-1.860	1503.7	1478.5	1300.9	1047.5	692.6	606.1	563.2	652.5	5583.7	5.04	0.0
36	-1.740	1522.8	1497.0	1315.9	1057.2	695.2	607.5	564.1	652.5	5587.5	5.05	0.0
37	-1.620	1540.9	1514.7	1330.1	1066.5	697.6	608.9	565.0	652.5	5591.3	5.06	0.0
38	-1.500	1558.1	1531.4	1343.5	1075.4	699.9	610.3	565.9	652.5	5595.1	5.06	0.0
39	-1.380	1574.3	1547.1	1356.2	1083.7	702.1	611.6	566.8	652.5	5598.9	5.07	0.0
40	-1.260	1589.4	1561.8	1368.1	1091.5	704.2	613.0	567.8	652.5	5602.7	5.07	0.0
41	-1.140	1603.4	1575.5	1379.1	1098.7	706.2	614.3	568.7	652.5	5606.6	5.08	0.0
42	-1.020	1616.3	1588.0	1389.2	1105.5	708.1	615.5	569.7	652.5	5610.4	5.08	0.0
43	-0.900	1628.0	1599.5	1398.5	1111.6	709.9	616.8	570.6	652.5	5614.3	5.09	0.0
44	-0.780	1638.6	1609.7	1406.8	1117.2	711.6	617.9	571.6	652.5	5618.1	5.10	0.0
45	-0.660	1647.9	1618.8	1414.2	1122.1	713.2	619.1	572.5	652.5	5621.9	5.10	0.0
46	-0.540	1656.0	1626.7	1420.7	1126.5	714.7	620.2	573.5	652.5	5625.7	5.11	0.0
47	-0.420	1662.8	1633.4	1426.1	1130.2	716.0	621.3	574.5	652.5	5629.5	5.11	0.0
48	-0.300	1668.4	1638.8	1430.6	1133.4	717.3	622.4	575.4	652.5	5633.2	5.12	0.0
49	-0.180	1672.6	1643.0	1434.1	1135.9	718.4	623.4	576.4	652.5	5636.9	5.13	0.0
50	-0.060	1675.6	1645.9	1436.6	1137.7	719.4	624.3	577.3	652.5	5640.6	5.13	0.0

51	0.060	1677.3	1647.5	1438.0	1139.0	720.3	625.3	578.3	652.5	5644.3	5.14	0.0
52	0.180	1677.6	1647.9	1438.5	1139.6	721.1	626.1	579.2	652.5	5647.8	5.15	0.0
53	0.300	1676.7	1647.0	1437.9	1139.5	721.7	627.0	580.2	652.5	5651.4	5.15	0.0
54	0.420	1674.4	1644.8	1436.4	1138.9	722.3	627.8	581.1	652.5	5654.9	5.16	0.0
55	0.540	1670.9	1641.4	1433.8	1137.5	722.7	628.5	582.1	652.5	5658.4	5.16	0.0
56	0.660	1666.0	1636.7	1430.3	1135.6	723.0	629.3	583.0	652.5	5661.8	5.17	0.0
57	0.780	1659.9	1630.8	1425.8	1133.0	723.2	629.9	583.9	652.5	5665.1	5.18	0.0
58	0.900	1652.6	1623.7	1420.3	1129.9	723.3	630.5	584.8	652.5	5668.4	5.18	0.0
59	1.020	1644.0	1615.3	1413.8	1126.1	723.3	631.1	585.8	652.5	5671.7	5.19	0.0
60	1.140	1634.1	1605.8	1406.4	1121.7	723.1	631.7	586.7	652.5	5674.8	5.20	0.0
61	1.260	1623.1	1595.1	1398.1	1116.7	722.8	632.2	587.6	652.5	5677.9	5.20	0.0
62	1.380	1611.0	1583.4	1388.9	1111.2	722.4	632.6	588.4	652.5	5680.9	5.21	0.0
63	1.500	1597.7	1570.5	1378.8	1105.1	721.9	633.0	589.3	652.5	5683.9	5.21	0.0
64	1.620	1583.3	1556.5	1367.8	1098.4	721.3	633.4	590.2	652.5	5686.8	5.22	0.0
65	1.740	1567.9	1541.5	1356.1	1091.3	720.6	633.7	591.0	652.5	5689.6	5.23	0.0
66	1.860	1551.5	1525.6	1343.5	1083.6	719.7	633.9	591.8	652.5	5692.4	5.23	0.0
67	1.980	1534.1	1508.7	1330.2	1075.4	718.7	634.1	592.7	652.4	5695.0	5.24	0.0
68	2.100	1515.7	1490.9	1316.2	1066.8	717.6	634.3	593.5	652.4	5697.6	5.24	0.0
69	2.220	1496.5	1472.3	1301.5	1057.7	716.5	634.4	594.3	652.4	5700.2	5.25	0.0
70	2.340	1476.5	1452.8	1286.2	1048.2	715.1	634.5	595.0	652.4	5702.6	5.26	0.0
71	2.460	1455.7	1432.6	1270.2	1038.3	713.7	634.5	595.8	652.4	5705.0	5.26	0.0
72	2.580	1434.2	1411.7	1253.7	1028.1	712.2	634.5	596.5	652.4	5707.3	5.27	0.0
73	2.700	1412.0	1390.2	1236.6	1017.4	710.5	634.4	597.3	652.4	5709.5	5.27	0.0
74	2.820	1389.2	1368.1	1219.1	1006.5	708.8	634.3	598.0	652.4	5711.6	5.28	0.0
75	2.940	1365.9	1345.4	1201.1	995.2	706.9	634.2	598.7	652.4	5713.7	5.28	0.0
76	3.060	1342.0	1322.2	1182.8	983.7	705.0	634.0	599.3	652.4	5715.6	5.29	0.0
77	3.180	1317.7	1298.6	1164.0	971.9	702.9	633.7	600.0	652.4	5717.5	5.29	0.0

78	3.300	1293.0	1274.6	1144.9	959.9	700.8	633.4	600.6	652.4	5719.4	5.30	0.0
79	3.420	1264.0	1246.4	1122.7	945.9	698.5	633.1	601.3	652.4	5721.1	5.30	0.0
80	3.540	1239.2	1222.3	1103.4	933.7	696.1	632.7	601.9	652.4	5722.8	5.31	0.0
81	3.660	1214.1	1197.9	1084.0	921.4	693.7	632.3	602.4	652.4	5724.4	5.31	0.0
82	3.780	1188.7	1173.2	1064.3	908.8	691.1	631.8	603.0	652.4	5725.9	5.32	0.0
83	3.900	1164.0	1149.2	1045.2	896.8	689.1	631.3	603.5	652.4	5727.4	5.32	0.0
84	4.020	1138.2	1124.1	1025.2	884.0	686.3	630.7	604.0	652.4	5728.8	5.33	0.0
85	4.140	1112.3	1099.0	1005.1	871.1	683.4	630.1	604.5	652.4	5730.1	5.33	0.0
86	4.260	1086.4	1073.8	984.9	858.1	680.5	629.5	605.0	652.4	5731.3	5.33	0.0
87	4.380	1060.4	1048.5	964.7	845.0	677.4	628.8	605.5	652.4	5732.5	5.34	0.0
88	4.500	1034.5	1023.3	944.4	831.9	674.3	628.0	605.9	652.4	5733.6	5.34	0.0
89	4.620	1008.6	998.1	924.2	818.8	671.1	627.3	606.3	652.4	5734.6	5.34	0.0
90	4.740	982.8	973.0	904.0	805.6	667.9	626.5	606.7	652.4	5735.6	5.35	0.0
91	4.860	957.1	948.0	883.9	792.5	664.5	625.6	607.0	652.4	5736.5	5.35	0.0
92	4.980	931.5	923.1	863.9	779.4	661.1	624.8	607.4	652.4	5737.3	5.35	0.0
93	5.100	906.1	898.3	844.0	766.3	657.6	623.8	607.7	652.4	5738.1	5.36	0.0
94	5.220	880.8	873.8	824.2	753.3	654.1	622.9	608.0	652.4	5738.8	5.36	0.0
95	5.340	855.8	849.4	804.5	740.3	650.5	621.9	608.3	652.4	5739.4	5.36	0.0
96	5.460	830.1	824.4	784.3	727.0	646.8	620.9	608.5	652.4	5740.0	5.36	0.0
97	5.580	805.9	800.8	765.2	714.3	643.1	619.8	608.7	652.4	5740.6	5.36	0.0
98	5.700	781.8	777.3	746.2	701.6	639.3	618.7	608.9	652.4	5741.0	5.37	0.0
99	5.820	757.9	754.1	727.3	689.0	635.5	617.6	609.1	652.4	5741.4	5.37	0.0
100	5.940	734.3	731.1	708.6	676.5	631.6	616.5	609.2	652.4	5741.8	5.37	0.0

EXIT PRESSURE	EXIT TEMPERATURE
2244.8	609.3

POWER PER UNIT LENGTH (KW/FT) 5.95

Example 8.

The effect of oversized cladding on PWR thermal-hydraulic performance is to be studied. Consider the PWR described in Example 7 but with an outer cladding diameter of 0.500 in. (instead of 0.422 in.). Compare the following thermal-hydraulic parameters for the "oversized cladding" and "normal" flow channels:

a. Location and magnitude of maximum fuel temperature

b. Location and magnitude of maximum cladding temperature

c. Exit water temperature.

Solution.

Each flow channel in the fuel assembly will experience the same pressure drop. The "oversized cladding" flow channel will have a smaller equivalent diameter and hence will pass less flow for the same pressure loss (as shown in Example 5). To assure approximately the same pressure loss as in the "normal" channels the mass velocity of the "oversized cladding" channel must be adjusted. The equivalent diameters are computed first.

"Normal" channel:

$$A_f = S^2 - \frac{\Pi}{4} D_s^2$$

$$= (0.563)^2 - \frac{\Pi}{4} (0.422)^2$$

$$= 0.1771 \text{ in.}^2$$

$$p_w = \Pi D_s = \Pi(0.422)$$

$$= 1.327 \text{ in.}$$

$$D_e = 4 A_f / p_w$$

$$= 4\ (0.1771)/1.327$$

$$= 0.534 \text{ in.}$$

"Oversized cladding" channel:

$$A_f = S^2 - \frac{\Pi}{4} D_s^2$$

$$= (0.563)^2 - \frac{\Pi}{4}(0.500)^2$$

$$= 0.1206 \text{ in.}^2$$

$$p_w = \Pi D_s = \Pi(0.500)$$

$$= 1.571 \text{ in.}$$

$$D_e = 4\ A_f/D_e$$

$$= 4(0.1206)/1.571$$

$$= 0.307 \text{ in.}$$

From the results of Example 5,

$$G_{oc} = G_N(D_{eoc}/D_{eN})^{2/3}$$

$$= 2.2 \times 10^6\ (0.307/0.534)^{2/3}$$

$$= 1.52 \times 10^6 \text{ lbm/hr-ft}^2.$$

The input to the PWR Thermal-Hydraulics Code is the same as that for Example 7 except for the new value of G computed above and the oversized D_s. From the code (printout next seven pages) the following results are obtained:

```
*****************************
 PWR THERMAL HYDRAULICS CODE
*****************************

 INPUT DATA

 FUEL O.D.        CLAD O.D.        ROD PITCH        ACTIVE CORE LENGTH
 ( INCH )         ( INCH )         ( INCH )             ( INCH )
  0.3664           0.5000           0.5630               144.00

 INLET MASS VELOCITY   CORE INLET TEMPERATURE   CORE INLET PRESSURE
   ( LB/HR-FT**2 )             ( F )               (    PSIA    )
     0.1520E 07               543.00                  2250.00

 VOLUMETRIC THERMAL    EXTRAPOLATED HEIGHT
  ( BTU/HR-FT**3 )          ( INCH )
     0.4000E 08              158.40

                 OPTION    1

 E N D   O F   I N P U T   D A T A
 ******************************


   AT THE INLET OF THE CORE

CHANNEL AVE. VELOCITY   REYNOLDS #   FRICTION FACTOR
     (FT/SEC)
       8.99              164104.4         0.0167

**************************************************************************
```

NO	Z-LOC.	TO	T1/4	T1/2	T3/4	TF	T CLAD	T BULK	T SAT	H T COEF	DELTA P	CHF RATIO
1	-5.940	684.9	681.8	660.2	629.3	586.1	551.9	543.2	652.6	3993.5	4.31	0.0
2	-5.820	712.3	708.7	682.8	645.8	594.1	553.9	543.6	652.6	3994.7	4.31	0.0
3	-5.700	740.2	735.9	705.7	662.6	602.2	555.9	544.0	652.6	3996.1	4.31	0.0
4	-5.580	768.5	763.6	729.0	679.5	610.3	558.0	544.5	652.6	3997.7	4.32	0.0
5	-5.460	796.6	791.0	751.9	696.0	617.8	560.1	545.1	652.6	3999.5	4.32	0.0
6	-5.340	825.5	819.3	775.6	713.1	625.7	562.2	545.7	652.6	4001.5	4.32	0.0
7	-5.220	856.4	849.4	800.7	731.1	633.6	564.4	546.4	652.6	4003.7	4.32	0.0
8	-5.100	886.4	878.8	825.2	748.6	641.5	566.6	547.1	652.6	4006.1	4.33	0.0
9	-4.980	916.9	908.6	850.0	766.4	649.3	568.9	547.9	652.6	4008.6	4.33	0.0
10	-4.860	947.7	938.7	875.1	784.2	657.1	571.2	548.8	652.6	4011.3	4.34	0.0
11	-4.740	978.9	969.1	900.4	802.2	664.8	573.6	549.7	652.6	4014.2	4.34	0.0
12	-4.620	1010.4	999.9	925.9	820.3	672.5	576.0	550.7	652.6	4017.3	4.34	0.0
13	-4.500	1042.1	1030.9	951.6	838.5	680.1	578.4	551.7	652.6	4020.5	4.35	0.0
14	-4.380	1074.2	1062.1	977.5	856.7	687.6	580.8	552.8	652.6	4023.9	4.35	0.0
15	-4.260	1106.4	1093.6	1003.6	875.0	695.1	583.3	553.9	652.6	4027.4	4.36	0.0
16	-4.140	1138.8	1125.2	1029.7	893.3	702.4	585.7	555.1	652.6	4031.1	4.36	0.0
17	-4.020	1171.4	1157.0	1055.9	911.7	709.7	588.2	556.3	652.6	4034.9	4.37	0.0
18	-3.900	1204.0	1188.8	1082.2	930.0	716.9	590.8	557.6	652.6	4038.9	4.38	0.0
19	-3.780	1236.6	1220.7	1108.5	948.3	724.0	593.3	558.9	652.6	4042.9	4.38	0.0
20	-3.660	1269.3	1252.5	1134.7	966.5	730.9	595.8	560.2	652.6	4047.1	4.39	0.0
21	-3.540	1301.8	1284.3	1160.8	984.6	737.8	598.4	[illegible]	652.6	4051.4	4.40	0.0
22	-3.420	1334.3	1315.9	1186.8	1002.5	744.5	600.9	563.1	652.6	4055.8	4.40	0.0

23	-3.300	1371.3	1352.0	1216.3	1022.5	751.1	603.5	564.5	652.6	4060.3	4.41	0.0
24	-3.180	1403.9	1383.8	1242.3	1040.4	757.6	606.1	566.1	652.6	4064.8	4.42	0.0
25	-3.060	1436.1	1415.2	1268.1	1058.0	764.0	608.6	567.6	652.6	4069.4	4.43	0.0
26	-2.940	1468.0	1446.3	1293.5	1075.5	770.2	611.2	569.2	652.6	4074.1	4.44	0.0
27	-2.820	1499.4	1476.9	1318.6	1092.6	776.3	613.8	570.8	652.6	4078.8	4.44	0.0
28	-2.700	1530.2	1507.0	1343.2	1109.5	782.2	616.3	572.4	652.6	4083.6	4.45	0.0
29	-2.580	1560.5	1536.4	1367.3	1125.9	788.0	618.8	574.1	652.6	4088.3	4.46	0.0
30	-2.460	1590.0	1565.2	1390.9	1142.0	793.6	621.4	575.8	652.6	4093.1	4.47	0.0
31	-2.340	1618.7	1593.2	1413.8	1157.7	799.0	623.9	577.5	652.6	4097.9	4.48	0.0
32	-2.220	1646.6	1620.4	1436.0	1172.8	804.3	626.4	579.3	652.6	4102.7	4.49	0.0
33	-2.100	1673.5	1646.7	1457.5	1187.5	809.5	628.8	581.0	652.6	4107.5	4.50	0.0
34	-1.980	1699.5	1672.0	1478.2	1201.7	814.4	631.3	582.8	652.5	4112.2	4.51	0.0
35	-1.860	1724.4	1696.2	1498.1	1215.2	819.2	633.7	584.6	652.5	4116.9	4.52	0.0
36	-1.740	1748.1	1719.4	1517.1	1228.2	823.8	636.1	586.4	652.5	4121.6	4.53	0.0
37	-1.620	1770.7	1741.4	1535.1	1240.6	828.3	638.5	588.2	652.5	4126.2	4.55	0.0
38	-1.500	1792.0	1762.2	1552.1	1252.3	832.5	640.8	590.0	652.5	4130.7	4.56	0.0
39	-1.380	1812.1	1781.7	1568.2	1263.4	836.6	643.1	591.9	652.5	4135.1	4.57	0.0
40	-1.260	1830.8	1800.0	1583.2	1273.7	840.5	645.4	593.7	652.5	4139.4	4.58	0.0
41	-1.140	1848.2	1816.9	1597.2	1283.4	844.1	647.6	595.6	652.5	4143.7	4.59	0.0
42	-1.020	1864.2	1832.5	1610.0	1292.4	847.6	649.8	597.4	652.5	4147.8	4.61	0.0
43	-0.900	1878.7	1846.8	1621.8	1300.6	850.9	651.9	599.3	652.5	4151.8	4.62	0.0

SUBCOOLED BOILING

44	-0.780	1894.8	1862.5	1635.0	1310.3	855.8	655.9	601.1	652.5	4015.2	4.63	0.18

SUBCOOLED BOILING

45 -0.060 1903.2 1870.6 1641.6 1314.5 856.7
655.9 603.0 652.5 4176.3 4.64 0.18

SUBCOOLED BOILING

46 -0.540 1910.2 1877.5 1647.1 1318.1 857.5
655.9 604.8 652.5 4345.4 4.66 0.18

SUBCOOLED BOILING

47 -0.420 1915.9 1883.0 1651.4 1320.9 858.1
655.9 606.7 652.5 4523.1 4.67 0.18

SUBCOOLED BOILING

48 -0.300 1920.1 1887.1 1654.7 1323.0 858.6
655.9 608.5 652.5 4710.3 4.68 0.19

SUBCOOLED BOILING

49 -0.180 1922.9 1889.8 1656.9 1324.4 858.9
655.9 610.4 652.5 4907.8 4.70 0.19

SUBCOOLED BOILING

50 -0.060 1924.3 1891.2 1658.0 1325.1 859.1
655.9 612.2 652.5 5116.7 4.71 0.19

SUBCOOLED BOILING

51 0.060 1924.3 1891.2 1658.0 1325.1 859.1
655.9 614.0 652.5 5338.1 4.72 0.19

SUBCOOLED BOILING

52 0.180 1922.9 1889.8 1656.9 1324.4 858.9
655.9 615.8 652.5 5573.5 4.74 0.19

SUBCOOLED BOILING

53 0.300 1920.1 1887.1 1654.7 1323.0 858.6
655.9 617.6 652.5 5824.1 4.75 0.20

SUBCOOLED BOILING

54 0.420 1915.8 1882.9 1651.4 1320.9 858.1
655.9 619.4 652.5 6092.1 4.76 0.20

SUBCOOLED BOILING

55 0.540 1910.2 1877.5 1647.0 1318.1 857.5
655.9 621.1 652.5 6379.3 4.78 0.20

SUBCOOLED BOILING

56 0.660 1903.2 1870.6 1641.5 1314.5 856.7
655.9 622.8 652.5 6688.2 4.79 0.20

SUBCOOLED BOILING

57 0.780 1894.7 1862.4 1635.0 1310.3 855.7
655.9 624.6 652.5 7021.8 4.81 0.20

SUBCOOLED BOILING

58 0.900 1884.9 1852.9 1627.4 1305.4 854.6
655.9 626.3 652.5 7383.5 4.82 0.21

SUBCOOLED BOILING

59 1.020 1873.8 1842.0 1618.7 1299.8 853.3
655.9 627.9 652.5 7777.3 4.83 0.21

SUBCOOLED BOILING

60 1.140 1861.3 1829.9 1608.9 1293.5 851.9
655.9 629.6 652.5 8208.2 4.85 0.21

SUBCOOLED BOILING

61 1.260 1847.5 1816.4 1598.2 1286.6 850.3
655.9 631.2 652.5 8682.7 4.86 0.21

SUBCOOLED BOILING

62 1.380 1832.3 1801.7 1586.4 1279.0 848.6
655.8 632.8 652.5 9207.8 4.87 0.21

SUBCOOLED BOILING

63 1.500 1816.0 1785.8 1573.6 1270.7 846.6
655.8 634.4 652.5 9793.3 4.89 0.21

SUBCOOLED BOILING

64 1.620 1798.3 1768.7 1559.9 1261.9 844.6
655.8 636.0 652.5 10451.2 4.90 0.22

SUBCOOLED BOILING

65	1.740	1779.5	1750.4	1545.2	1252.4	842.4
655.8	637.5	652.5	11196.7	4.92	0.22	

SUBCOOLED BOILING

66	1.860	1759.5	1730.9	1529.7	1242.3	840.0
655.8	639.0	652.5	12050.6	4.93	0.22	

SUBCOOLED BOILING

67	1.980	1738.4	1710.4	1513.2	1231.7	837.5
655.8	640.5	652.5	13039.4	4.94	0.22	

SUBCOOLED BOILING

68	2.100	1716.3	1688.9	1495.9	1220.5	834.9
655.8	641.9	652.5	14200.1	4.96	0.23	

SUBCOOLED BOILING

69	2.220	1693.1	1666.3	1477.8	1208.8	832.1
655.8	643.4	652.5	15585.5	4.97	0.23	

SUBCOOLED BOILING

70	2.340	1668.9	1642.8	1459.0	1196.6	829.1
655.7	644.7	652.5	17270.8	4.98	0.23	

SUBCOOLED BOILING

71	2.460	1643.9	1618.4	1439.4	1183.9	826.1
655.7	646.1	652.5	19370.3	5.00	0.24	

SUBCOOLED BOILING

72	2.580	1618.0	1593.2	1419.2	1170.7	822.8
655.7	647.4	652.5	22064.8	5.01	0.25	

SUBCOOLED BOILING

73	2.700	1591.3	1567.3	1398.3	1157.1	819.5
655.7	648.7	652.5	25658.2	5.02	0.26	

SUBCOOLED BOILING

```
74    2.820      1563.9      1540.6      1376.9      1143.2       816.0
 655.7      650.0       652.5    30707.1        5.03        0.28
```

SUBCOOLED BOILING

```
75    2.940      1535.8      1513.3      1355.0      1128.9       812.4
 655.6      651.2       652.5    38341.8        5.05        0.31
```

NET VOID FORMATION

1. Subcooled (surface) boiling begins in the coolant channel at Z = -0.780 ft; i.e., about 3/4 foot upstream of the center of the core. No boiling was observed for the normal channel.
2. The coolant bulk temperature exceeds the local saturation temperature at about 3 ft past the center of the core. The calculations were terminated at this point since this is an improper operating condition for the PWR. The reason for the observed behavior is that the "oversized cladding" coolant channel is flow starved and can not remove the energy supplied to it by the fuel without boiling.
3. The maximum fuel temperature is 1924.3 F and occurs at the center of the core. This compares to the 1677.6 F at Z = 0.180 ft for normal core operation.
4. The maximum cladding temperature observed was 859.1 F and also occurred near the center of the core. This compares to 723.3 F at Z = 0.960 ft for normal operation.

5.6 <u>Listing of Code</u>

```
C
C
C
C--------VPI PWR THERMAL HYDRAULICS CODE.
C
C
C
      INTEGER OP
      REAL KFU,KCL,KBC,M,NUCIRC,NU,MDOT,KCL1,KCL2,KFU1,KFU2
C
C        TEMPEPATURE DEPENDENT PROPERTIES.
C
      KFU(T)=0.61021E1-0.46365E-2*T+0.13063E-5*T**2
      KCL(T)=0.53643E1+0.23214E-2*T
      KBC(T)=0.11711+0.13910E-2*T-0.18102E-5*T**2
      VIS(T)=0.85067-0.17501E-2*T+0.11420E-5*T**2
      RHO(T)=0.5778E2+0.28018E-1*T-0.88346E-4*T**2
      CP(T)=0.47088E1-0.15753E-1*T+0.17233E-4*T**2
C
C        SATURATION TEMPERATURE AS FUNCTION OF PRESSURE.
C
      TSAT(P)=0.45602E3+0.11033*P-0.10199E-4*P**2
C
C        PRESSURE DEPENDENT "CONSTANTS" FOR EQUATION N-64.
C
      M(P)=0.48141E-1+0.22471E-3*P+0.83018E-9*P**2
      C(P)=+0.11312E1-0.78004E-3*P+0.33119E-6*P**2-0.56355E-10*P**3
C
C        READING INPUT DATA.
C
      READ(5,5)DFU,DCL,S,H
    5 FORMAT(3F10.4,F10.1)
```

```
      READ(5,10)GIN,TIN,PIN
   10 FORMAT(E10.3,2F10.1)
      READ(5,15)QTPO,HE,OP
   15 FORMAT(E10.3,F10.1,I10)
C
C         PRINTOUT OF INPUT DATA.
C
      WRITE(6,20)
   20 FORMAT(1H1,1X,'******************************',/,3X,'PWR THERMAL HY
     <DRAULICS CODE',/,2X,'******************************',//,3X,'INPUT D
     <ATA',//)
      WRITF(6,25)DFU,DCL,S,H
   25 FORMAT(3X,'FUEL O.D.',6X,'CLAD O.D.',6X,'ROD PITCH',6X,'ACTIVE COR
     <E LENGTH',/,3X,'( INCH )',7X,'( INCH )',7X,'( INCH )',12X,'( INCH
     <   )',/,4X,F6.4,9X,F6.4,9X,F6.4,14X,F6.2,/)
      WRITE(6,30)GIN,TIN,PIN
   30 FORMAT(3X,'INLET MASS VELOCITY',3X,'CORE INLET TEMPERATURE',3X,'CO
     <RE INLET PRESSURE',/,5X,'( LB/HR-FT**2 )',13X,'( F )',15X,'(   PSI
     <A   )',/,7X,E11.4,14X,F7.2,17X,F7.2,/)
      WRITE(6,35)QTPO,HE
   35 FORMAT(3X,'VOLUMETRIC THERMAL',4X,'EXTRAPOLATED HEIGHT',/,4X,'( BT
     <U/HR-FT**3 )',11X,'( INCH )',/,6X,E11.4,15X,F6.2,//)
      WRITE(6,40)OP
   40 FORMAT(19X,'OPTION',3X,I2,///,3X,'E N D  O F  I N P U T  D A T A',
     </,3X,'******************************',////)
C
C         CONVERTING INCHES TO FEET.
C
      DFU=DFU/12.0
      S=S/12.0
      DCL=DCL/12.0
      H=H/12.0
```

```
      HE=HE/12.0
      PI=3.14159
C
C        GEOMETRY CALCULATIONS.
C
      AFL=S**2.0-PI*DCL**2.0/4.0
      PW=PI*DCL
      DEQ=4.0*AFL/PW
      SRAT=S/DCL
C
C        RATIO OF ACTUAL FRICTION FACTOR OF COOLANT CHANNEL TO CIRCU
C        TUBE FRICTION FACTOR.
C
       FRAT=-3.4751+8.0528*SRAT-4.7047*SRAT**2+0.91621*SRAT**3
C
C        MASS FLOW RATE CALCULATION.
C
      MDOT=GIN*AFL
C
C        SELECTION OF AXIAL CALCULATION INCREMENT.
C
      DELL=H/100.0
C
C        CALCULATION OF PRESSURE LOSS AT INLET OF CORE.
C
      UB=GIN/RHO(TIN)/3600.0
      RE=UB*DEQ*RHO(TIN)/VIS(TIN)*3600.0
      FCIRC=0.184/RE**0.2
      F=FCIRC*FRAT
      DELPF=F*DELL*RHO(TIN)*UB**2/DEQ/2.0/32.2
      DELPE=0.5*RHO(TIN)*UB**2/2.0/32.2
C
```

```
C          CALCULATION OF CONDITIONS AT BEGINNING OF FIRST INCREMENT.
C
      P1=PIN-(DELPF+DELPE)/144.0
      TB1=TIN
      I=1
      CHFR=0.0
C
C          CALCULATION OF AXIAL COORDINATE OF CENTER OF FIRST INCREMENT.
C
      ZC=-H/2.0+DELL/2.0
C
C          CALCULATION OF AVERAGE BULK TEMPERATURE FOR INCREMENT.
C
      WRITE(6,42)
   42 FORMAT(5X,'AT THE INLET OF THE CORE',//)
      WRITE(6,45)UB,RE,FCIRC
   45 FORMAT(1X,'CHANNEL AVE. VELOCITY ',2X,'REYNOLDS #',3X,'FRICTION FA
     <CTOR',/,7X,'(FT/SEC)',/,8X,F6.2,12X,F8.1,8X,F7.4,//)
      WRITE(6,50)
   50 FORMAT('***********************************************************
     <****************************************************************
     <****',///)
      WRITE(6,55)
   55 FORMAT(3X,'NO',3X,'Z-LOC.',6X,'TO',7X,'T1/4',6X,'T1/2',6X,'T3/4',7
     <X,'TF',6X,'T CLAD',4X,'T BULK',5X,'T SAT',3X,'H T COEF',4X,'DELTA
     <P',3X,'CHF RATIO',/)
   60 QTPAV=QTPO*COS(PI*ZC/HE)
      TB2=TB1+QTPAV*PI*DFU**2*DELL/4.0/MDOT/CP(TB1)
      TBAV=(TB1+TB2)/2.0
C
C          CALCULATION OF PRESSURE DROP AND AVERAGE PRESSURE FOR INCREMENT
C
```

```
      UB=GIN/RHO(TBAV)/3600.0
      RE=UB*DEQ*RHO(TBAV)/VIS(TBAV)*3600.0
      FCIRC=0.184/RE**0.2
      F=FCIRC*FRAT
      DELPF=F*DELL*RHO(TBAV)*UB**2.0/DEQ/2.0/32.2
      P2=P1-DELPF/144.0
      PAV=(P1+P2)/2.0
C
C        CALCULATION OF AVERAGE SATURATION TEMPERATURE FOR INCREMENT.
C
      TSATAV=TSAT(PAV)
C
C        CALCULATION OF HEAT TRANSFER COEFFICIENT FOR INCREMENT.
C
      PR=CP(TBAV)*VIS(TBAV)/KBC(TBAV)
      NUCIRC=0.023*PR**0.4*RE**0.8
      NU=NUCIRC*FRAT
      HTC=NU*KBC(TBAV)/DEQ
C
C        CALCULATION OF TEMPERATURE AT OUTSIDE SURFACE OF CLADDING FOR
C        INCREMENT.
C
      TSAV=TBAV+QTPAV*DFU**2.0/4.0/DCL/HTC
C
C        CHECK FOR POSSIBILITY OF NET VOID FORMATION.  IF DETECTED, "NE
C        VOID FORMATION" PRINTED OUT AND CALCULATIONS TERMINATED.
C
      IF(TB2-TSATAV)75,65,65
   65 WRITE(6,70)
   70 FORMAT(/,10X,'  NET VOID FORMATION',//)
      GO TO 180
C
```

```
C          CHECK FOR POSSIBILITY OF SURFACE BOILING.  IF DETECTED, "SUB-
C          COOLED BOILING" PRINTED OUT AND BOILING BURNOUT CHECK INITIATED
C
   75 IF(TSAV-TSATAV)115,80,80
   80 WRITE(6,85)
   85 FORMAT(/,10X,'   SUBCOOLED BOILING',/)
C
C          CALCULATION OF CRITICAL HEAT FLUX RATIO.
C
      CHF=C(PAV)*(GIN/10.0**6)**M(PAV)*(TSATAV-TBAV)**0.22*10.0**6
      SHFAV=QTPAV*DFU**2.0/4.0/DCL
      CHFR=SHFAV/CHF
C
C          CHECK FOR BOILING BURNOUT.  IF DETECTED, "SUBCOOLED BURNOUT"
C          PRINTED OUT AND CALCULATIONS TERMINATED.
C
      IF(SHFAV-CHF)110,90,90
   90 WRITE(6,95)
   95 FORMAT(/,10X,'   SUBCOOLED BURNOUT',/)
      WRITE(6,100)
  100 FORMAT(10X,'CHF',20X,'SHF',/)
      WRITE(6,105)CHF,SHFAV
  105 FORMAT(6X,E10.3,6X,E10.3)
      GO TO 180
C
C          RECALCULATION OF TEMPERATURE AT OUTSIDE OF SURFACE CLADDING AND
C          HEAT TRANSFER COEFFICIENT FOR SUBCOOLED BOILING CASES.
C
  110 TSAV=TSATAV+1.9*SHFAV**0.25*EXP(-PAV/900.0)
      HTC=SHFAV/(TSAV-TBAV)
C
C          CALCULATION OF TEMPERATURE AT OUTSIDE SURFACE OF FUEL FOR IN-
```

```
C        CREMENT.  DETERMINED BY ADDING TEMPERATURE RISE ACROSS CLADDING
C        TO TEMPERATURE AT OUTSIDE SURFACE OF CLADDING.  AN ITERATION
C        TECHNIQUE IS USED TO OBTAIN THE CLADDING THERMAL CONDUCTIVITY
C        TO WITHIN 1%.
C
  115 KCL1=KCL(TSAV)
  120 TF=TSAV+QTPAV*DFU**2.0/8.0/KCL1*ALOG(DCL/DFU)
      TCLAV=(TSAV+TF)/2.0
      KCL2=KCL(TCLAV)
      ERR=ABS((KCL1-KCL2)/KCL2)
      IF(ERR-0.010)130,130,125
  125 KCL1=KCL2
      GO TO 120
C
C        CALCULATION OF TEMPERATURE AT CENTER OF FUEL FOR INCREMENT.  AN
C        ITERATION TECHNIQUE IS USED TO OBTAIN FUEL THERMAL CONDUCTIVITY
C        WITHIN 1%.
C
  130 KFU1=KFU(TF)
  135 TO=TF+QTPAV*DFU**2.0/16.0/KFU1
      TFAV=(TF+TO)/2.0
      KFU2=KFU(TFAV)
      ERR=ABS((KFU1-KFU2)/2.0)
      IF(ERR-0.010)145,145,140
  140 KFU1=KFU2
      GO TO 135
C
C        CALCULATION OF FUEL TEMPERATURE AT 1/4, 1/2, AND 3/4 FUEL RADII
C        FOR INCREMENT.
C
  145 T14=TO-QTPAV*DFU**2.2/16.0/KFU1/16.0
       T12=TO-QTPAV*DFU**2.0/16.0/KFU1/4.0
```

```
      T34=TO-QTPAV*CFU**2.0/16.0/KFU1/16.0*9.0
C
C        PRINTOUT OF RESULT FOR CALCULATION INCREMENT.
C
      WRITE(6,150)I,ZC,TO,T14,T12,T34,TF,TSAV,TBAV,TSATAV,HTC,DELPF,CHFR
  150 FORMAT(1X,I3,F10.3,9F10.1,4X,F6.2,6X,F6.2)
C
C        CHECK TO SEE IF CALCULATIONS HAVE BEEN COMPLETED FOR ENTIRE
C        CORE.
C
      IF(I-100)155,160,160
C
C        SETTING CONDITIONS FOR BEGINNING OF NEXT CALCULATION INCREMENT.
C
  155 I=I+1
      ZC=ZC+DELL
      TB1=TB2
      P1=P2
C
C        TRANSFERRING TO START CALCULATIONS FOR NEXT INCREMENT.
C
      GO TO 60
C
C        CALCULATION OF EXIT PRESSURE LOSS .
C
  160 UB=GIN/RHO(TB2)/3600.0
      RE=UB*DEQ*RHO(TB2)/VIS(TB2)*3600.0
      FCIRC=0.184/RE**0.2
      F=FCIRC*FRAT
      DELPF=F*DELL*RHO(TB2)*UB**2/DEQ/2.0/32.2*6.0
      DELPF=RHO(TB2)*UB**2/2.0/32.2
C
```

```
      CALCULATION OF CORE EXIT CONDITIONS OF COOLANT.

   PEX=P2-(DELPF+DELPE)/144.0
   TEX=TB2

      PRINTOUT OF CORE EXIT COOLANT CONDITIONS.

   WRITE(6,165)
65 FORMAT(//,10X,'EXIT PRESSURE',10X,'EXIT TEMPERATURE',/)
   WRITE(6,170)PEX,TEX
70 FORMAT(13X,F6.1,18X,F6.1)

      CALCULATION OF AVERAGE POWER OF FUEL ROD IN KW/FT.
      PRINTOUT OF AVERAGE POWER.

   POVL=QTPO*HE*DFU**2/2.0*SIN(PI*H/2.0/HE)/H
   POVL=POVL/3412
   WRITE(6,175)POVL
75 FORMAT(//,5X,' POWER PER UNIT LENGTH (KW/FT)  ',F6.2,//
60 STOP
   END
```

6.0 References

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7.0 Problems

1. By performing an energy balance on a differential element with sides dx, dy, and dz, derive the steady state heat conduction equation in Cartesian coordinates.

2. Derive the steady state heat conduction equation in cylindrical coordinates.

3. Develop an expression for the temperature distribution in a long, thin, hollow cylindrical fuel element with inner radius r_i, outer radius r_o, and volumetric thermal source strength q'''. Assume that the inner surface is insulated and at temperature T_i.

4. A long, thin, hollow cylindrical fuel element with volumetric thermal source strength q''' is cooled on both the inner and outer surfaces ($r = r_i$ and $r = r_o$ surfaces) such that the surface temperatures are T_i and T_o, respectively. Derive an expression for the location of the maximum temperature in the fuel.

5. Calculate the ratio of peak power to average power for a fuel rod of constant cross section assuming that the volumetric thermal source strength varies as $q''' = q_o''' \cos (\Pi Z/H_e)$ and that the extrapolation length is 10% of the core height. Compare this ratio to that for the case where the extrapolation length is assumed to be zero.

6. Show that for a cylindrical shell without internal heat generation the use of an average constant thermal conductivity evaluated at the arithmetic mean of the inner and outer surface temperatures is equivalent to a linear variation of thermal conductivity with temperature.

7. A PWR fuel rod consists of 0.500 in. diameter UO_2 fuel pellets in a 0.500 in. ID, 0.600 in. OD tube of Zircaloy 2 cladding. The heat transfer coefficient at the outer surface of the fuel rod is 5000 Btu/hr-ft^2-F. The thermal conductivities of the fuel and cladding are 1.95 Btu/hr-ft-F and 8.0 Btu/hr-ft-F, respectively. The coolant bulk temperature is 550 F and the volumetric thermal source strength is 1.2×10^8 Btu/hr-ft^3. Determine the maximum fuel temperature and the maximum cladding temperature.

8. Same as Problem 7 except the ID of the cladding is 0.540 in. The 0.040 in. diametrical gap between the fuel and cladding is filled with helium gas. Assuming that the heat transfer through the helium is by molecular conduction only (k_{he} = 0.140 Btu/hr-ft-F) determine the maximum fuel and cladding temperature.

9. A ten foot long PWR fuel assembly contains 0.520 in. OD fuel rods spaced 0.680 in. in a square array. The average conditions in the core are 2500 psia and 550 F. The mass velocity of the water is 2.5×10^6 lbm/hr-ft^2. Determine the pressure drop through the core.

10. Determine the effect on pressure drop of a 10% increase in cladding OD for the fuel in Problem 9. Assume the spacing, coolant conditions and mass velocity remain the same.

11. Determine the heat transfer coefficient for the fuel and flow conditions of Problem 9.

12. A PWR fuel consists of 0.50 in. diameter UO_2 fuel pellets contained in 0.50 in. ID, 0.57 in. OD Zirconium 2 cladding. The fuel rods are spaced 0.70 in. in a square array. At a particular location in the core the pressure is 2500 psia, the temperature is 520 F and the coolant velocity is 16 ft/sec. The volumetric thermal source strength is 6×10^7 Btu/hr-ft^3. Determine the heat transfer coefficient.

13. Determine the coolant velocity required to keep the temperature rise across the convective layer below 65 F for the fuel and coolant conditions of Problem 12.

14. In a nuclear fuel rod the volumetric thermal source strength varies axially as $q'''(Z) = q_o''' \cos(\Pi Z/H_e)$. The energy generated in this rod is removed by a coolant channel which has a mass flow rate of $\dot{m}$ and an inlet temperature of T_{in}. Designate the fuel diameter as D_f and the cladding OD as D_s. Develop an expression for the axial variation of the temperature at the inside surface of the cladding. Your expression should contain only T_{in}, q_o''', D_f, D_s, H, H_e, $\dot{m}$, C_p, h, k_c, and Z.

15. From the expression developed in Problem 14 determine the location of the maximum cladding temperature in the reactor.

16. Determine the location of the maximum fuel temperature for the fuel rod - coolant channel system of Problem 14.

17. A 2760 Mw(t) PWR contains 31,000 fuel rods each composed of 0.380 in. diameter UO_2 fuel pellets in 0.440 in. OD tubes of Zircaloy - 4 cladding. The fuel rods are spaced 0.580 in. in a square array. The reactor core height is 150 in. The coolant inlet conditions are 2250 psia, 553 F and the average mass velocity is 2.7×10^6 lbm/hr-ft^2. The thermal source strength varies as $q''' = q_o''' \cos(\Pi Z/H_e)$ and $H_e = 165$ in. Determine the maximum fuel temperature and the maximum cladding temperature in the core.

18. Determine the % reduction in coolant flow that will lead to incipient surface boiling for the reactor described in Problem 17.

19. Determine the % overpower that the reactor described in Problem 17 can produce without net steam formation. What fraction of the core experiences surface boiling at this limiting case?

20. A 2450 Mw(t) PWR contains 37,000 fuel rods with an active fuel length of 153 in. Each fuel rod consists of 0.362 in. diameter UO_2 pellets enclosed in 0.420 in. OD Zircaloy - 4 cladding. The fuel rods are spaced 0.558 in. in a square array. The inlet coolant conditions are 455 F, 2500 psia and the average velocity in the core is 13.8 ft/sec. The axial power distribution is a truncated cosine function. Assume an extrapolation length of 8 in. Determine the location and magnitude of the maximum fuel, cladding, and coolant temperatures.

21. Determine the % of overpower that would result in incipient surface boiling in the core described in Problem 20. At what location does this occur?

22. One of the coolant channels in the core described in Problem 20 is obstructed such that there is a 30% decrease in coolant flow. When and where does surface boiling occur? Does net void formation occur? If so, where?